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July 19, 1984

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Snyder:

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

Attached for your information is the updated Technical Evaluation Report (TER) for the Submerged Demineralizer System (SDS). This update reflects the current operating configuration and limitations of the SDS. For example, Appendix 1, "Reactor Coolant Processing Plan", has been modified to incorporate Supplement No. 1 to Appendix 1, "Reactor Coolant Processing with the RCS in a Partially Drained Condition". Additionally, this update incorporates the NRC's comments stated in NRC Letter NRC/TMI 83-067 from L. H. Barrett to B. K. Kanga dated November 8, 1983. These comments and GPU Nuclear's response are attached for your reference.

If you have any questions concerning this information, please call Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

BKK/RDW/jep

Attachment

cc: Acting Deputy Program Director - TMI Program Office, Mr. P. J. Grant

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RESPONSE TO NRC COMMENTS ON THE SDS TER

NRC Comment No. 1

TER, Main Section, Chapter 6, Page 72, First Paragraph:

The words "ingestion of contaminated foods" should be added as has been done in the corresponding statements of the appendices.

GPU Nuclear Response No. 1

Chapter 6 has been completely revised to reflect off-site radiological exposures based on current SDS operations. The section that the above comment references has been deleted.

NRC Comment No. 2

TER, Main Section, Chapter 7

Some additional numerical data should be added:

a. Section 7.1, Page 80 and Section 7.2, Page 82: What percentage of the limits specified in 10 CFR 20 does the airborne contamination present?

b. Section 7.5, Page 87: What would be the collective dose to the personnel and to which level the airborne activity would be reduced?

GPU NUclear Response No. 2a

Sections 7.1 and 7.2 have been revised to delete the reference to 10 CFR 20. Instead, a table has been provided on Page 86 which lists the release parameters for various isotopes.

GPU Nuclear Response No. 2b

Section 7.5 has not been revised since the original NRC approval of the SDS TER. With respect to the specific comments, the first part of the question is hypothetical and a response cannot be accurately provided. The question is dependent upon the amount of personnel who were in the area and the amount of airborne radioactivity released. With respect to the second part of the question, this issue was previously addressed in the SDS Safety Evaluation Report (SER) (NUREG-0796) which concluded that the off-site radiological consequences for a cask drop accident are well within the guidelines of 10 CFR 100.

NRC Comment No. 3

TER, Main Section, Chapter 7, Section 7.5, Page 87

The mentioned studies about plant operability and integrity after the postulated incident should be given precise references.

GPU Nuclear Response No. 3

Section 7.5 does not mention studies about plant operability and integrity after the postulated incident, i.e., Inadvertent Drop of SDS Shipping Cask. The effects on systems, components, and structures which are described in this section are based on the original safety evaluation of the SDS.

NRC Comment No. 4

TER Appendix 1

a. Chapter 3, Figure 3.4 does not show the Sandpiper Pump and associated piping referred to in Section 3.3, Page 19.

b. Chapter 4, Section 4.2.1, Page 32, figure for the volume of Reactor Coolant System water processed has been left blank.

GPU Nuclear Response No. 4

Appendix 1, "Reactor Coolant Processing Plan", has been replaced with Supplement 1 to Appendix 1, "Reactor Coolant Processing Plan with the Reactor Coolant System in a Partially Drained Condition", which reflects current system operations. Figure 3.4, "RCS Processing Collection Stage", has been renumbered as Figure 3.3 and includes the Sandpiper Pump, RCF-P1. The figure shows that this pump is normally disconnected. Section 1.1 of Appendix 1 states the volume of water that has been processed from the RCS, i.e., in excess of 700,000 gallons.

NRC Comment No. 5

TER, Appendix 2, Chapter 3, Section 3.1.1, Page 9

The description of the solid waste handling system omits liner inertization (LRVOS) which is referred to in the corresponding section of the main part (Section 5.5, Page 59).

GPU Nuclear Response No. 5

The above comment has been incorporated into Section 3.1.1.

NRC Comment No. 6

TER, Main Section, Chapter 4, Section 4.2, Page 37

On the second line, the words "A fuel pool" should be deleted.

GPU Nuclear Response No. 6

Comment accepted; reference Page 37.

NRC Comment No. 7

TER, Main Section, Chapter 6, Section 6.2.3.1, Page 65

The words "Automatic level controlled" do not reflect the new situation. Vent and drain subsystem operating procedures now call for automatic "high level indication", but manual startup of the bottom pump, which then stops automatically at low level. Section 5.1.6, Page 55, gives better information.

GPU Nuclear Response No. 7

Comment accepted. The term "Automatic" has been changed to "Manual".

NRC Comment No. 8

TER, Appendix 1, Chapter 3, Section 3.4.2, Page 22

Reference to the feed tank standpipe should be deleted. The off-gas separator bottoms are routed directly to the RCS cleanup manifold from which they can be sent to the Miscellaneous Waste Holdup Tank (MWHT) or Reactor Coolant Bleed Tanks (RCBT's) for storage, or to the SDS for processing.

GPU Nuclear Response No. 8

Comment accepted: reference Section 3.4.2 of Appendix 1 (revised).

NRC Comment No. 9

Two sets of data about activity concentrations in SDS feed are found in the reports. The first one is based on February 1982 samples, the high values recorded then reflect the situation before any SDS operation. The second one is based on April 1983 RCS samples; the rather low values measured then are the results of one year decontamination efforts of the RCS water.

The main part of the TER uses the first set only. Chapter 1, giving a kind of historical review of SDS operation, should mention both sets and include some additional data about RB decon water; it is not clear why current sample data were not added to Table 1.1 to reflect current expected operating conditions. In chapter 3, which discusses expected performance of SDS and EPICOR ion exchange systems, use of the old data (Table 3.1) is questionable. Since the radionuclide concentrations of water to be processed now will be much lower than during the first runs, the DF's to be obtained will presumably be lower than those registered in early 1982. We do not doubt that radionuclide concentrations will be reduced; however, because of the variation in water sources, flow rates, and process system configuration additional discussions should be provided on predicted performance of the ion exchangers. The same comments apply for Chapter 4, Section 4.3.3. In Chapter 6 (Table 6.1) use of the higher values is acceptable since it adds to the conservatism of the radiological safety study.

Appendix 1 uses both data sets and is therefore inconsistent (old data in Table 1.1; the new data in Table 4.1). Furthermore, the new data are used for the radiological safety analysis, the conservatism of which is so decreased.

Supplement 1 to Appendix 1 and Appendix 2 use the second set only and so are internally consistent.

GPU Nuclear Response No. 9

Table 1.1 of the main section has been updated to reflect recent activity levels of SDS processed water. This is consistent with the intent of reflecting current system operations. Table 3.1, Effluent Concentrations of February 1982, has been clarified to indicate the data is for historical purposes (see Section 3.5, first paragraph). Additionally, Table 3.2 has been added which reflects effluent concentrations as of June 1984. Section 4.3.3 has been revised to reference Table 3.2. Table 6.1 has been revised to reflect a history of concentrations of the radiologically significant isotopes (Cs-137 and Cs-134). Table 1.1 of Appendix 1 has been updated and is consistent with Table 1.1 of the main section. Table 4.1 of Appendix 1 has been deleted. Tables 1.1 and 4.1 of Appendix 2 have been updated and are consistent with the data in the main section and Appendix 1.

NRC Comment No. 10

The methodology for estimating the ^{129}I contribution to the population dose is not the same in the main part of the TER and in the appendices.

The main part reads (Page 70): "However, no further effluent treatment is assumed for (...) ^{129}I. (...) For ^{129}I the DF is 1". This leads to a thyroid dose commitment of 20 mrem/Ci ^{129}I from the SDS for the critical location (using figures in pp. 75 and 78).

Appendix 1 reads (Page 35): "A decontamination factor of 100 is assumed for (...) iodine in the plant Waste Gas System (...). It is further assumed that (...) iodine pass(es) through HEPA filters in place at TMI-2 to give an additional DF of 100 (...). Therefore, the total DF for (...) iodine including both the plant Waste Gas System and treatment previously existing at

TMI-2 is 10^4. This leads to a thyroid dose commitment of 0.02 mrem/Ci ^{129}I from SDS for the critical location (using figures in pp 36 and 38).

Supplement 1 to Appendix 1 and Appendix 2 copy the statements in Appendix 1 without modification.

GPU Nuclear Response No. 10

Section 6.3.2 of the main section, "Off-site Radiological Exposures", to which the first part of the above comment refers, has been completely revised to reflect exposure rates based on current SDS processing. ^{129}I has been deleted from this chapter since its level in SDS processed water is virtually negligible, i.e., less than 10^{-8} µCi/ml prior to RCS processing. This value is less than that stated in the SDS SER (NUREG 0796 dated June 1981) which evaluated concentration of ^{129}I of 5.7×10^{-6} µCi/mP..

NRC Comment No. 11

In the main part, the contaminated gas stream is assumed to be the effluent of the SDS Vent and Drain Subsystem with a gas flow rate of 650 cfm. In the appendices this gas stream is not referred to and the contaminated gas stream is assumed to be the effluent of the RCBT's vents through the plant Waste Gas System, with a gas flow rate of 0.67 cfm. HEPA filters do not allow any credit for iodine decontamination; the presence and location of charcoal filters, if any, should be clearly mentioned in the various systems.

GPU Nuclear Response No. 11

The above comment concerning the previous revision of the SDS TER refers to the following sections: 6.3.2 of the main part (Page 69), 4.2.2 of Appendix 1 (Page 35), and 4.2.2 of Appendix 2 (Page 29). These analyses referenced different release paths for gaseous effluents from the SDS due to different processing flow paths. As stated in GPU Nuclear's response to NRC Comment No. 10, the remaining iodine activity is negligible and is not included in the revised gaseous effluent source term analysis of Section 6.3.2 of the main body of the SDS TER, therefore, charcoal filters are not required.

NRC Comment No. 12

In Appendix 1, the total volume of RCS water to be treated is estimated to be 350,000 gallons (Page 3), however, the off-site dose estimation is based on a volume roughly equal to that of the RCS (12 days of continuous operation at 5 gpm or 88,000 gallons, Page 35). Your basis for these calculations appears marginal: indeed, for the nuclides which are trapped in the SDS liners ^{134}Cs, ^{137}Cs, ^{90}Sr) it neglects the recontamination of RCS water by such mechanisms as steady leaching of fuel debris in the reactor core, and for the nulides which are not affected by the process, like ^{3}H, ^{85}Kr, ^{125}Sb, ^{129}I, it is obvious that a constant concentration is to be considered during the whole RCS feed and bleed process. Actually, it is known that the given RCS processing target (given as 1 μCi/ml ^{137}Cs in the report, but since then brought down to 0.1 μCi/ml) can be reached and hold only if the feed and bleed is continued regularly.

The same comments apply to Appendix 2 (Page 30) where the Appendix 1 statements have been copied without further evaluation.

GPU Nuclear Response No. 12

Section 1.3 of Appendix 1, to which the above comment refers, and the data referred to in Appendix 1 have been deleted. Off-site dose estimation is described in Chapter 6 in the main section.

NRC Comment No. 13

In the main part of the TER, Chapter 1, Section 1.4.2, provide references to procedures describing the ratios of various zeolites to be used in the ion exchanger mix (Page 13) and the alternative processing modes (Page 15).

In chapter 8, Sections 8.2 and 8.3 (Page 91) the text should clearly state who is qualified to approve SDS testing.

GPU Nuclear Response No. 13

Section 1.4.2 has been revised to reference Operating Procedure 2104-8.14 (reference Page 12). Section 8.2 has been revised and states, "The SDS system testing will be approved by the GPU Nuclear Startup and Test Manager". It is not necessary to include this statement in Section 8.3.

NRC Comment No. 14

At several points in Appendices 1 and 2 a chloride concentration of 5 ppm is given as an SDS operating target. Beyond the fact that SDS processing is without effect on chloride concentration, we should point that the value of 5 ppm is a Technical Specification limit and that therefore the operating target should be lower. (Appendix 1, Section 2.3 and Appendix 2, Section 2.3). We should like to have more detail about the Recovery Operations Plan Change Request referred to at the end of Appendix 1, Section 2.3 (Page 11).

GPU Nuclear Response No. 14

Section 2.3 of Appendices 1 and 2 have been revised to indicate that chloride concentration will be maintained at less than 5 ppm. NRC Amendment of Order dated April 24, 1984, increased the maximum boron concentration to 6000 ppm and limited the Ph of the reactor coolant from 7.5 to 8.4.

NRC Comment No. 15

The criticality issue is discussed in Appendix 1 Chapter 3, Section 3.4.5.3 (Page 25). Identify whether any additional sampling is planned for specific TRU detection in addition to the spectroscopic equipment used at the RCBT's for monitoring of ^{144}Ce/Pr.

GPU Nuclear Response No. 15

The above section has been revised to state "Filter and Process Train Samples are being Analyzed for Isotopes for Uranium and Plutonium". GPU Nuclear is removing the TRU monitoring system from service. The purpose of the TRU monitoring system was to detect the presence of fuel that has been transported into the process stream during letdown into the Reactor Coolant Bleed Tanks and collected there. Since the start of RCS processing, approximately 800,000 gallons of water have been letdown into the RCBT-C without a significant increase in count rate, indicating that there has been no transportation of fuel into the process stream during letdown. Recent readings from the TRU's have decreased by a factor of 10 to 100 (Reference 5), which is more evidence that fuel is not accumulating in the RCBT-C. Furthermore, the letdown of the last batch of reactor coolant prior to head lift has been completed. After head lift, reactor coolant processing will normally take place using the IIF Processing System. According to the IIF Processing SER (Reference 6), this processing system "...tranfers water from the IIF through the SDS to a Reactor Coolant Holdup Tank," which is either the A or the C bleed tank with the other tank being used concurrently for makeup to the reactor vessel. In summary, during IIF processing, the RCBT-C will not have unfiltered reactor coolant being pumped into it. Further, past reactor coolant processing history indicates that normal letdown does not transport significant quantities of fuel. Therefore, significant accumulation in RCBT-C is not considered possible.

NRC Comment No. 16

There is some inconsistency between Appendix 1, Chapter 5, Section 5.2 (Page 40) and main part of the TER, Chapter 4, Section 4.3.7.10 (Page 43). The test pressure is said to be 1.5 times the design pressure in the main part, and 1.1 times in the appendix, which does not clearly tell if the figure refers to pneumatic or hydrostatic testing. Code ANSI B31.1 asks for testing at 1.5 times the design pressure.

For piping systems and components added after 1980, the 1980 revision of ANSI B31.1 should be referred to (main part, Chapter 4, Section 4.3.6.1).

GPU Nuclear Response No. 16

Section 5.2 of Appendix 1 has been deleted. Section 4.3.6.1 of the main part has been revised to reference ANSI B31.1.

NRC Comment No. 17

In Appendix 1, Chapter 2, Section 2.2.2, the Standby Pressure Control System is erroneously said to be described in the FSAR.

GPU Nuclear Response No. 17

Section 2.2.2 of Appendix 1 has been revised to indicate that the Standby Pressure Control System is described in the TMI-2 Recovery System Descriptions and Technical Evaluation Reports.

NRC Comment No. 18

We should like to have a clear statement about when and for what purpose sand filters will be used in the place of zeolite liners in the "cation" positions of the SDS. We understand, that the replacement occurred during RCS draindown, but the statements in the report are somewhat confusing (SD Chapter 3, Section 3.4.1, Pages 87-88: TER main part Chapter 1, Section 1.4.2, Page 14: Chapter 4, Section 4.3.7.2b, Page 42: Chapter 5, Section 5.1.2, Page 52: TER Appendix A, Chapter 3, Section 3.4.2, Page 22). If the SDS configuration has to be changed when switching from RCS or IIF water treatment to RB decon water treatment, will there be sufficient time for that operation? What will be the RB sump processing requirement during those periods?

GPU Nuclear Response No. 18

The above sections have been revised to reference the sand filters, i.e., "cation", which have been installed in place of the post filters which were jumpered. The filters will not have to be changed when switching from either RCS or IIF to RB decon water treatment. The RB sump will be processed through EPICOR 2 which will be addressed in the next update of the EPICOR 2 System Description.

NRC Comment No. 19

In regard to the IIF level control system, Appendix 2 of the TER, Chapter 6, Section 6.2.1 (Page 38) and Appendix 17 of the System Description, Section 4.2.1, describe switch failure analysis and contingencies, including any mitigation plans about RCS overflow in the Reactor Building.

GPU Nuclear Response No. 19

Section 6.2.1 of Appendix 2 has been revised to indicate that Valve MU-V9 which controls the flow rate to the IIF may be manually controlled. Also, when initially filling the IIF, the water level will be visually monitored to prevent overfilling of the IIF.

NRC Comment No. 20

In the main part of the TER, Chapter 7, Section 7.5 (Page 87), an assumed airborne release of 10^{-4} percent (i.e., 10^{-6}) does not seem consistent with the 0.01 percent used in the SDS SER (NUREG-0796, Section 5.2). Section 7.5 should be reviewed with such a basis. In our evaluation, the final conclusion would not be affected.

GPU Nuclear Response No. 20

Section 7.5 has been revised to conform with Section 5.2 of SDS SER (i.e., a factor of 10^{-4}). As your comment noted, this change does not affect the final conclusion.

GPU Nuclear

T E R 3527-006 REV 2

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DIVISION

TECHNICAL EVALUATION REPORT

FOR

Submerged Demineralizer System

GPU Nuclear TECHNICAL EVALUATION REPORT

No. 3527-006

Title: SUBMERGED DEMINERALIZER SYSTEM (SDS)

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Initial issue per GPUNC Letter 4400-82-L-0066		4/82
1	Reissue per GPUNC Letter 4410-83-L-0122		6/83
2	Reissue per GPUNC Letter 4410-84-L-0109 Incorporates changes required by ECM's S-1151 (Revisions 0 through 3), S-1163 (Revisions 0 through 3), 1110 Revision 0, 1140 Revision 0, 1159 Revision 0, and 1141 Revision 0.		7/84

REVISED TER
7/84

TECHNICAL EVALUATION REPORT

SUBMERGED DEMINERALIZATION SYSTEM

JULY 1984

CONTENTS

Chapter 1

Summary of Treatment Plan

1.1 Project Scope

To date the SDS system has processed in excess of 2 million gallons of contaminated water, including: 650,000 gallons of Reactor Building sump water, 366,000 gallons from RB decon and 760,000 gallons of RCS water. The continued decontamination of TMI-2 includes the repeated processing of the IIF/RCS. The activity level of this water is given in Table 1.1. In addition, Reactor Building Decon water or water from other sources may be processed through SDS as necessary.

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

1. Placement of the operating system in the spent fuel pool to take advantage of shielding provided by the water in the pool.

2. Radioactive gas collection and treatment prior to release.

3. Liquid leak-off collection and treatment.

4. Underwater placement of ion-exchange vessels into a shipping cask without removal from the spent fuel pool.

5. Use of existing EPICOR-II equipment for polishing of SDS effluent, as required.

1.2 Identification of Radionuclides and Radioactivity Levels

Water samples were taken from the reactor coolant system and the containment sump, and were analyzed to identify specific radionuclides and concentrations. Typical results are listed in Table 1.1. The Reactor Coolant System (RCS) and containment sump specific radionuclides and concentrations are based upon actual sample data taken. The RCS activity decreases due to radioactive decay and leakage from the RCS. However, RCS activity may increase during processing shutdown due to leaching.

1.3 Alternatives Considered

During the early phases of developing a system for the control, clean-up, and disposition of the contaminated water located in the containment building of TMI-2, several methods or alternatives were evaluated. These alternatives were grouped into two categories:

(1) those with no volume reduction, and
(2) those with volume reduction.

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

<u>Alternative I</u>: Leave Contaminated Water in Containment Indefinitely (No Volume Reduction)

<u>Discussion</u>

A. <u>Containment Sump Water</u>

1. The sump water contains radionuclide concentrations as depicted in Table 1.1. The existence of this may cause some increase in radiological exposure problems during the recovery program, i.e., increased exposure to recovery program personnel, increased contamination levels, and increased possibility of airborne radioactivity.

2. The presence of the contaminated sump water would prevent decontamination of the lower levels of the containment building.

B. Reactor Coolant System Water

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

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

1. The potential for increased personnel exposure exists. Therefore, compliance with the principles of ALARA is not possible.

2. Facility decontamination and defueling operations are seriously inhibited or perhaps prevented.

3. Continued storage of the contaminated water in the containment sump for increased periods of time increases the probability that leakage from the building may occur. Leakage of contaminated water from the reactor building sump may threaten the public health and safety.

4. Continued storage of the water in the containment building for an extended period of time is undesirable. The primary isotopes of concern (Cs-137 and Sr-90) exhibit decay half-lives of approximately 30 year. Storage in the containment sump for approximately 300 years would be required for 10 half-life decay.

Maintenance of containment integrity for this interval of time cannot be assured.

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

Discussion:

1. To safely contain the contaminated water, the construction of an on-site liquid radwaste storage facility would be required.

2. Additional radiation areas on the plant site would be created if a liquid radwaste storage facility were built.

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

4. A liquid radioactive waste transfer system for the transfer of the contaminated water from the various locations to the waste storage complex would be required.

5. Handling and pumping operations may involve leakage and the spread of contamination.

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

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

<u>Alternative III</u>: Solidification and Disposal (No Volume Reduction)

<u>Discussion</u>:

1. The construction of an on-site solidification facility would be required.

2. Based on 1,000,000 gallons of contaminated water originally to be processed, a 30-gallon availability of water volume in a 55-gallon drum, 70% availability, 24-hour/day operation, and a 45 minute cycle time, the processing time may exceed four years.

3. Based on 1,000,000 gallons of contaminated water originally to be processed and a 30-gallon availability of water volume in a 55-gallon drum. The number of drums of solidified waste that would be generated would exceed 33,000. Handling, transportation and disposal of this extremely large quantity of solidified waste would be prohibitively expensive and violate basic principles of minimizing radioactive waste volumes.

4. The handling evolution required to solidify the contaminated water may involve substantial radiation exposure to personnel.

5. The potential for leakage and contamination problems may be substantial in operating a solidification facility for processing this contaminated water in this manner.

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

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

Discussion:

1. The system would be capable of concentrating fission products on a medium to effectively remove those products from the water.

2. Processing contaminated water would result in concentrated waste requiring additional shielding.

3. The system incorporates remote operability features.

4. Design, construction and operation would allow for relatively short lead times.

5. The system would require minimal maintenance.

6. The SDS is amenable to location within the Spent Fuel Pool which would utilize the shielding capability of the pool water.

7. Containers of highly loaded ion exchange media arising from operation of the SDS would not be acceptable at shallow land disposal sites. The SDS design and selection of ion exchange media allows volumes of such highly loaded media to be minimized to permit interim storage and probable ultimate disposal in a geological repository. It is believed that the EPICOR-II liners, generated as a result of polishing the SDS effluent, will be suitable for shallow land disposal because of their low curie content.

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

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

Alternative V: Evaporation (Volume Reduction)

Discussion:

1. Evaporation would require the design and construction of a new facility.

2. Due to the nature of the contaminated water to be processed the design of the facility would be complex to allow for maintenance of the processing system and personnel radiological protection. The construction of the facility may require at least four years.

3. Evaporation provides the ability to process a wide range of chemical contaminants.

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

1.4 Description of the Decontamination Process

1.4.1 General

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

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

Of the volume reduction category, the Submerged Demineralizer system (SDS) in conjunction with EPICOR II for final polishing, or Alternative IV, was chosen as the most appropriate process for the following reasons:

1. Basic design simplicity.

2. High performance for decontaminating liquids, i.e., decontamination factors up to 10^7, or higher.

3. Amenable to placement under water to take advantage of shielding properties of the water

4. Ability to implement water processing in a timely fashion for support of the overall objective of fuel removal.

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

The SDS with EPICOR II is an ion-exchange process expected to provide decontamination factors of up to 10^7 for cesium and 10^6 for strontium (see Table 3.1), thus removing the

majority of the activity from the water prior to placement in the Processed Water Storage Tanks, or usage for continued decontamination or makeup to the RCS.

1.4.2 SDS Operating Description

Figure 1.1 shows a block diagram of the process flow of the Submerged Demineralizer System (SDS) with the EPICOR II System. Radioactive water enters the SDS via the RCS manifold. This source of water can pass through two cartridge or sand type filters for removal of particulate matter.

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

The first part of the SDS ion-exchange system consists of up to six underwater vessels (24 1/2 in. x 54 1/2 in.). Each vessel contains approximately 8 cubic feet of homogeneously mixed IE-96 and LINDE-A zeolite ion exchange media. Zeolite media volumes and mixtures may be changed to reflect different processing scenarios (The resin mix is specified by Radiochemical Engineering on the form included in OP 2104-8.14). Inlet, outlet, and vent connections are made with remotely operated couplings. The vessels are arranged in two parallel trains with three columns in each train. Flow may be directed through one train of three vessels or through both trains in parallel. Loading of the vessels will be controlled by feed batch size, residence time, influent and effluent sample analysis, and continuous monitoring.

The second part of the SDS ion exchange system consists of two parallel sand filter vessels located underwater and immediately downstream of the zeolite beds. These sand filters will contain a mixture of sand and are intended to remove system effluent particulates, primarily zeolite fines. The columns are intended to be operated singly.

Present SDS operations are envisioned to provide for radionuclide loading of the zeolite media to a maximum of 60,000 Ci of ^{134}Cs and ^{137}Cs at the time of shipping.

This loading level is based on restrictions imposed based on the shielding provided by the Chem-Nuclear 1-13C II shipping cask. From the point of view of minimizing waste volume generation it is desireable to load the zeolites to these higher levels.

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

Additionally some processing operations will require fewer than three (3) ion exchange units per train to achieve desired decontamination factors. In these cases jumpers will be installed to bypass the unused positions.

When the SDS is processing contaminated sump water, the effluent from the "cation" sand filters can be sent to EPICOR-II for polishing. When processing reactor coolant the effluent may be routed to installed tankage for injection back into the Reactor Coolant System as a source of makeup or to EPICOR for polishing. The spent ion-exchangers and filters of SDS will be retained under water in the spent fuel pool until removed. To transport spent ion-exchangers, they will be bulk dewatered, vacuum dewatered, and catalyst recombiner added, and loaded into shielded casks while under water and removed from the spent fuel pool. Following decontamination of the cask surface, the cask can then be loaded onto a trailer for transportation.

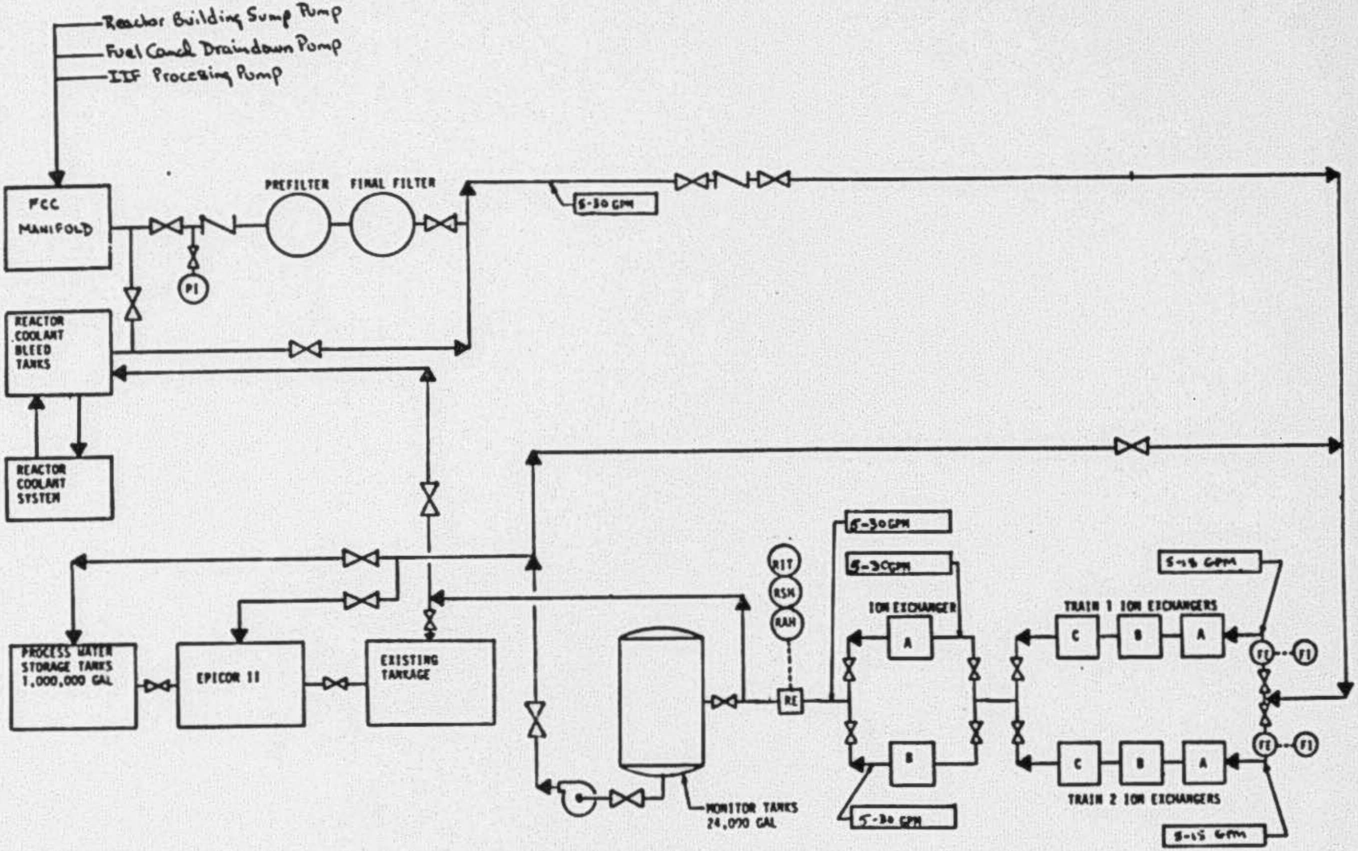

FIGURE 1.1
RCS PROCESSING FLOW SHEET

TABLE 1.1

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

Isotope	Reactor Coolant System	RB Sump Decon
	Radionuclide Concentrations (uCi/ml)	
Sampling Date	(6/84)	(7/83)
H3	0.045	0.12
Sr-90	2.7	1.6
Sb-125	0.14	0.023
Cs-134	0.015	0.14
Cs-137	0.3	2.1
pH	7.72	7
Boron	5200 ppm	3193 ppm
Na	1500 ppm	240 ppm

Chapter 2

Summary of Health and Environmental Effects

2.1 Occupational Radiation Exposure During Routine Operation

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

- o Shielding has been designed to limit whole body dose rates in operating areas to less than 1 mRem/hr. The filters and ion-exchangers are located approximately 16 feet underwater for shielding. Components and piping carrying high activity water not contained underwater in the fuel pool have been provided with shielding to maintain external dose rates to acceptable levels.

- o Controls and instrumentation are located in low ratiation areas.

- o Components containing high activity water have been designed for venting exhaust gases to the SDS Off Gas System. The Off-Gas System will minimize the potential for excessive airborne radioactivity releases in the work areas and to the environment.

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

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

2.1.1 Exposure Planning

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

- Review of operating, maintenance and surveillance procedures to assure precautions and prerequisites are adequate.

- Review of the installed system to identify potential problems during operation and the implementation of corrective actions.

- Operational evaluations during preoperational testing and system training will be performed to update exposure estimates.

- o Determination of radiation dose rates during normal operations and maintenance evolutions will be performed.

As these reviews are completed, operating and surveillance frequencies can be established; total occupational exposures can be updated for the various activities during SDS operation. This exercise will permit review of those activities estimated to yield the highest man-rem expenditure. Pre-examination to assure that every reasonable effort is expended to minimize personnel exposure may include the following considerations:

- o Reduction of the frequency of operation
- o Temporary or additional shielding
- o Tool modifications
- o Procedure modification
- o Personnel training to reduce work time
- o Component modifications

2.2 Exposures to the Public During Routine Operation of the SDS and EPICOR-II

Refer to Chapter 6 for information on exposures to the public from routine operation of the SDS and EPICOR-II processing.

2.3 Evaluation of Unexpected Occurrences

The radiological assessment of unexpected occurrences includes the analysis of five hypothetical accidents that are postulated to occur during operation of the system.

The first accident is an inadvertent pumping of RCS water into the fuel storage pool until a total of 225 gallons of radioactive water is released to the pool. No exposures occur to the public since the contaminated water is contained in the pool. The maximum exposure rate at a distance of six feet above the pool surface is estimated to be 4.2 mR/hour. Since the release of water occurs underwater, no significant internal exposures are expected for workers. The primary impact of the accident is the contamination of water in the Spent Fuel Pool (233,000 gallons). (Refer to Section 7.1)

The second hypothetical accident assumes a pipe is ruptured and RCS water is sprayed into the building and fuel storage pool. It is possible that workers could be contaminated, however, prompt implementation of emergency procedures would minimize radiation exposures. The radioactive materials would be contained within the building except small amounts of radionuclides that would become airborne and subsequently be released through the monitored station discharge. This airborne radionuclide release would not result in significant exposures to the public. (Refer to Section 7.2)

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

The fourth hypothetical accident evaluated considers the inadvertant raising of a loaded zeolite ion exchanger above the pool surface. The dose rate at a distance of 20 feet from the source is estimated to be approximately 340 Rem/hr. (Refer to Section 7.4)

The final hypothetical accident considers the inadvertant drop of the SDS shipping cask containing a loaded zeolite ion exchanger. The SDS shipping cask is assumed to be dropped from the maximum height of the Fuel Handling Building crane to the EL 305' floor. The dose rate resulting from a complete rupture of the SDS shipping cask at a distance of 20 feet is approximately 340 Rem/hr and assumes rupture of both the cask and the vessel. The small amounts of radionuclides assumed to become airborne would not result in significant exposures to the public. Also there would not be a significant effect from direct radiation exposure to the public. (Refer to Section 7.5) Evaluation of additional unexpected occurrences is covered in Appendix 2 to this TER.

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

2.4 Industrial Health and Safety

2.4.1 Public Safety

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

1. Lifting and handling activities described take place within the TMI complex.

2. Hazardous chemical species, flammable or explosive substances, heavy industrial processes, and concentrated manufacturing activities are not involved in the installation or operation of the SDS.

3. No toxic substances are used in the SDS.

2.4.2 Occupational Safety

During the operation of the SDS, operating personnel will adhere to station requirements for occupational safety. Structural equipment and operating equipment used shall meet Occupational Safety and Health Administration requirements as applicable. Personnel protective equipment that would be required for the operation of the SDS will be utilized in accordance with standard station procedures.

2.5 Non-Radiological Environmental Effects

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

2.6 Ultimate Waste Disposition

Radioactive material generated as a result of the accident at TMI is currently restricted to disposal at the commercial disposal site operated by U.S. Ecology at Hanford, Washington. SDS vessels meeting the criteria for disposal at this site will be disposed of by shallow land burial at this location. SDS vessels not meeting the Hanford Site criteria will be classified as abnormal waste and disposed of by the Department of Energy in accordance with the Memorandum of Understanding dated July 15, 1981, between the Nuclear Regulatory Commission and the Department of Energy dealing with the disposition of solid nuclear waste from the cleanup of TMI Unit 2.

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Chapter 3

Process Description

3.1 Introduction

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

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

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

After filtration, radioactive ion removal from the waste water involves the use of ion-exchange materials. The two or three ion-exchange columns (per train) contain homogeneously mixed inorganic zeolite material which effectively removes essentially all of the cesium and much of the strontium. Other trace levels of radionuclides are also partially removed by the zeolite media. The radioactivity content in the effluent stream of each bed is used to determine when the bed is expended and replaced.

Final demineralization of the contaminated sump water and selected batches of RCS water is intended to be by the EPICOR-II system. Essentially, all remaining radionuclides excluding tritium are expected to be removed from the water during this process step.

3.2 Ion-Exchange Concepts

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

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

The decontamination factor (DF) is the ratio of the concentration in the influent stream to that in the effluent stream and is used for determining the efficiency of a purification process for radionuclide removal.

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

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

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

n = Fraction of Q used

E_w = Equivalent weight of the nuclide under consideration

C_r = Nuclide concentration (weight/volume)

V = Feed throughput (number of ion-exchange bed volumes)

K = Unit conversion constant

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

Specifications which have been defined for this purification process include:

(1) The flow rate to provide an acceptable residence time for ion diffusion and exchange to occur.

(2) The cross-sectional area of the ion-exchange media to provide an acceptable linear velocity through the bed.

(3) The bed depth to result in an acceptable pressure drop.

(4) A uniform flow distribution and a uniform media distribution to reduce the potential for channeling.

(5) The ion-exchange media bead size to minimize atrition and large pressure drops.

(6) The curie loading to satisfy personnel exposure, radiation damage, transportation, and storage regulations.

(7) The cation form and the amount of ion-exchange media impurities to maximize removal of specific nuclides.

3.3 Ion-Exchange Materials

The ion-exchanger media selected for use in this processing system are an inorganic zeolite material that is commercially available and known as Ion Siv IE-96 (Na^+ form of IE-95), and LINDE-A, to be used for SDS

and cation and anion resins to be used in EPICOR II.

Zeolites are aluminosilicates with framework structures enclosing large and uniform cavities. Because of their narrow, rigid, and uniform pore size, they can also act as "molecular sieves" to sorb small molecules, but to exclude molecules that are larger than the opening in the crystal framework.

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

Organic ion exchange resins are typically gels and are classified as cross-linked polyelectrolytes. Their framework, or matrix, consists of an irregular, macromolecular, three-dimensional network of hydrocarbon chains. In cation exchangers, the matrix carries ionic groups such as SO_3^-, COO^-, $(PO_2)_1^=$, and in anion exchangers groups such as NH_4^+, Na^+, H^+ are carried. The framework of the organic resins,

in contrast to that of the zeolites, is a flexible random network which is elastic, can be expanded, and is made insoluble by introduction of cross-links which interconnect the various hydrocarbon chains. The extent of crosslinking establishes the mesh width of the matrix and, thus, the degree of swelling and the ion mobilities within the resin. This, in turn, determines the ion exchange rates and electric conductivity of the resin.

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

3.4 Resin Selection Criteria

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

(1) Exchange capacity
(2) Swelling equilibrium
(3) Degree of crosslinking
(4) Resin particle size
(5) Ionic selectivity
(6) Ion-exchange kinetics
(7) Chemical, radiolytic and physical stability
(8) Previous demonstrated performance (EPICOR-II)

Experimental studies with reactor coolant water have been conducted to support and verify the selection of these ion-exchangers; refer to ORNL TM-7448. Further, onsite studies have been performed to support and verify selection of the ion-exchange media. The decontamination factors for the major contaminants were measured using a number of candidate ion exchangers including the organic resins, HCR-S and SBR-OH, and the zeolite ION SIV IE-96 and LINDE-A. The results indicated the most favorable type of ion exchange media to be used in the cleanup process were the available cation-anion resins in combination with the zeolite exchanger.

Furthermore, as a result of processing in excess of 2,000,000 gallons of radioactively contaminated water from the Auxiliary Building, Reactor Buildin and RCS, we are confident that the SDS, with EPICOR-II used as a polishing system for treatment of SDS effluent, will continue to provide an effective means to decontaminate the contaminated waters. EPICOR-II resin loadings may be altered to improve polishing effectiveness, if required.

3.5 Predicted Performance of Ion-Exchangers

The concentrations of radionuclides in samples of water from the Reactor Coolant System have been measured. Those radionuclides still detectable in June, 1984 include Sr-90, Cs-134, Cs-137, and Sb-125. The expected performance of the SDS ion-exchangers, and the EPICOR-II ion exchangers is shown in Table 3.2. The concentrations of strontium and cesium are expected to be significantly reduced by processing through the SDS and EPICOR-II system. Table 3.1 is included to provide historical data on Reactor Building Sump water processing.

Antimony is expected to pass through the SDS ion exchangers and will end up as the predominant gamma emitter in the solution entering the EPICOR-II system. The Concentration of Sb-125 in the containment building sump sample is approximately 0.011 microcuries per milliliter.

3.6 Monitoring of Ion Exchangers

Methods which may be used to monitor the effectivenes of the ion exchangers include liquid sampling and in-line radiation detectors. Liquid samples of feed and effluent streams can also be used to establish the approximate curie loadings in the loaded beds.

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

Historical - RB Sump Processing

Nuclide	Effluent concentrations,[a] uCi/ml.						Effluent
			Zeolite columns				
	Feed	Filter	First	Second	Third	Fourth	EPICOR-II
^{3}H	0.88	0.88	0.88	0.88	0.88	0.88	0.88
^{60}Co	b	b	2E-5	2E-5	2E-5	2E-5	2.3E-6
^{90}Sr	5.02	5.02	2.5	1.0E-1	8.5E-3	5E-3	<1.0E-5
^{106}Ru	b	b	4.0E-4	4.0E-4	4.0E-4	4.0E-4	1E-6
^{125}Sb	b	b	1.1E-2	1.1E-2	1.1E-2	1.1E-2	3.4E-7
^{134}Cs	1.39E+1	1.39E+1	1.7E+0	1.1E-4	1.1E-4	1.1E-4	2E-8
^{137}Cs	1.23E+2	1.23E+2	1.5E+1	1.0E-3	1.0E-3	1.0E-3	2E-7
^{144}Ce	b	b	4.0E-4	4.0E-4	4.0E-4	4.0E-4	1E-6

[a] In uCi/ml as of February 1982 based on actual samples
[b] Not quantifiable by gamma spectroscopy due to overall sample activities.

TABLE 3.2
Actual activity concentrations[a] in SDS process streams after 200 bed volumes through each zeolite bed
(Based on continuous flow through two zeolite columns)

RCS Processing

	Effluent concentrations,[a] uCi/ml.				
			Zeolite columns		
Nuclide	Feed	Filter	First	Second	Sand Filter
^{60}Co	<2.0E-3	2.2E-3	1.2E-3	<1.6E-4	<2E-4
90_{Sr}	3.4	3.1	0.084	2.8E-3	3.0E-3
106_{Ru}	2.3E-2	<2E-2	<5.2E-3	<1.5E-3	<1.7E-3
125_{Sb}	0.16	0.15	0.15	0.14	0.15
134_{Cs}	0.025	0.023	1.2E-3	<1.1E-4	<1.2E-4
137_{Cs}	0.56	0.51	3.0E-2	<1.7E-4	<1.6E-4
144_{Ce}	<1.2E-2	<1.2E-2	<4.5E-3	<1.8E-3	<2.0E-3

[a] In uCi/ml as of June 1984 based on actual samples
[b] Not quantifiable by gamma spectroscopy due to overall sample activities.

Submerged Demineralizer System Design Basis

4.1 Introduction

The Submerged Demineralization System (SDS) is an underwater ion-exchange system which has been specifically designed to process higher-level waste waters*, with inherent system features for reduction of occupational and environmental exposures. The SDS is submerged in the spent fuel pool (1) to provide shielding during operation, (2) to permit access to the system during demineralizer changeout, (3) to minimize the hazard from potential accidents, and (4) to utilize an existing Seismic Category I facility. In conjunction with the SDS, the EPICOR-II system may be used to provide final polishing of the SDS effluent water for removal of trace quantities of radionuclides.

Design features for SDS include:

1. A prefilter and final filter in series, followed by two parallel trains of 2 or 3 zeolite ion-exchangers in series. These ion-exchangers are followed by two "cation" sand filters in parallel followed by the EPICOR-II equipment. This combination of filters and ion-exchangers achieves the desired process flow rates and decontamination factors (DF's).

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

2. Series operation logic that allows for sequencing the demineralization units to prevent activity breakthrough in the final zeolite bed and maximize activity loading on spent beds to accomplish the best possible activity concentration.

The design objectives are as follows:

a. A totally integrated system that is as independent as possible from existing waste systems at the Three Mile Island plant. The SDS is a temporary system for the recovery of TMI-2.

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

c. A system that could be operated with a minimum of exposure to personnel and a negligible risk to the public.

d. A system that could accomplish the objective listed above in a timely and cost effective manner.

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

4.2 Components of the SDS Waste Processing System

The SDS is comprised of the following components, all of which will be located in the Unit 2 B fuel pool, or in the near vicinity of the B fuel pool. (See Figure 5.6, General Layout Plan.)

1. Feed filtering system;

2. Two parallel ion exchange trains, each comprised of two or three 10-cubic-foot vessels loaded with 8 cubic-feet (nominal) of homogeneously mixed IE-96 and LINDE-A zeolite exchange media;

3. Two parallel "cation" sand filters containing graded sand filter media;

4. A monitoring and sampling system for control of demineralizer unit loading;

5. A secondary containment system for the filters and zeolite beds and radiation shielding for piping, valves, sampling, and monitoring systems;

6. Two monitoring tanks for collecting treated water.

7. An off-gas system for treating and filtering gases and vent air from the system;

8. A Liner Recombiner and Vacuum Outgassing System (LRVOS) designed to eliminate the potential of a combustible hydrogen and oxygen mixture existing in the SDS liners.

9. Associated piping, valving, and structural supports required for placement of system components;

10. Auxiliary systems including underwater ion-exchange column storage, a dewatering system, and analytical equipment;

11. Vent system to allow for venting of stored vessels.

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

4.3 Submerged Demineralizer System Design Criteria

4.3.1 Design Basis

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

- U.S. Nuclear Regulatory Guide 1.140 dated March, 1978
- U.S. Nuclear Regulatory Guide 1.143 dated July, 1978

- U.S. Nuclear Regulatory Guide 8.8, dated June, 1978

- U.S. Nuclear Regulatory Guide 8.10, dated May, 1977

- U.S. Nuclear Regulatory Guide 1.21 Revision 1, June 1974

- Code of Federal Regulations, 10 CFR 20, Standard for Protection Against Radiation

- Code of Federal Regulations, 10 CFR 50, Licensing of Production and Utilization Facilities.

4.3.2 Process

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

4.3.3 Performance

The isotopic inventory for the water to be processed is summarized in Table 1.1. The SDS followed by the EPICOR-II system~~s~~ is designed and operated such as to reduce the average isotopic specific activity of the treated waste streams. The expected performance of these systems is given in Table 3.2.

4.3.4 Capacity

Flow Rate - 5 to 30 GPM (up to 15 GPM per train). The system will have the ability to operate continuously; (subject to periodic maintenance shutdown)

4.3.5 Performance and Design Requirements

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

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

4.3.6 Piping System (piping, valves and pumps)

1. The mechanical and structural design criteria and fabrication of piping systems and piping components are specified in ANSI B31.1, 1977 Edition with Addendum through Winter 1978 or ANSI B31.1, 1980 for components added after 1980, and Table 1 of Regulatory Guide 1.143.

2. Piping system design shall be based on a maximum of 150 psi at 100°F.

3. Piping runs are generally designed to permit water flushing.

4. Instrument connections to piping systems are located to provide clearance for attachment, operation and maintenance.

4.3.7 Vessels and Tanks

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

2. The vessels shall be of two types:

   a. Primary ion-exchangers shall contain approximately eight (8) cubic feet of zeolite ion exchange media for the purpose of removing cesium and strontium from the waste water. Should our processing scenario be changed it may be necessary to alter the volume of the zeolite media. Should changes occur, the NRC will be informed.

b. Influent and "cation" sand filter units are planned to contain cartridge type filter assemblies or sand capable of removing particles greater than approximately 10 microns. SDS effluent filter capability has been provided to incorporate the capability to filter out ion-exchange media fines from the process stream should fines carryover occur.

3. The SDS ion-exchangers and filters shall be capable of functioning submerged under approximately 16 feet of water within the spent fuel pool.

4. The ion-exchangers shall be designed for 15 GPM nominal process rate, filters shall be designed for 50 GPM nominal; volume velocity through the loaded ion-exchangers shall be limited to prevent channeling or breakthrough.

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

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

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

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

9. Design Condition

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

   b. The following additional design conditions have been imposed:

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

10. Testing

The vessels shall be hydrostatically tested at 1.5 times the design pressure per ASME Section VIII.

4.3.8 Shielding Design

The shielding shall be designed to reduce levels resulting from the SDS to less than 1mR/hr, general area. The shielding for the EPICOR-II equipment is adequate for the processing of the SDS effluent because the SDS effluent water activity will be lower than the activity level of the water for which EPICOR-II shielding was originally designed.

4.3.9 Leakage

To minimize the operational impact of activity that can potentially leak from bad process connections to Fuel Pool B, SDS vessels are contained in secondary containment enclosures. Pool water is continuously drawn through these enclosures and passed through separate ion exchangers (Leakage containment). This design prevents the pool water from eventually attaining high level concentrations of radionuclides. Monitoring of potential leakage is accomplished through the established SDS Sampling System.

4.3.10 Building and Auxiliary Service Interfaces

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

1. All components of the SDS located in the Fuel Handling Building do not exceed the normal load capacities of the cranes in this area. The Fuel Handling Building auxiliary and main cranes have capacities of 15 tons and 110 tons, respectively.

2. The SDS will operate in the ambient conditions of the Fuel Handling Building as supplied by the building heating, ventilating and air conditioning system, and lighting system.

3. Auxiliary services supplied to the SDS are from the Demineralizer Water, Electrical Distribution, Instrument Air and Service Air Systems.

4. During installation of the system, no equipment was permanently attached to the fuel pool liner and no penetrations were made in the fuel pool liner.

5. Structural support for the system will be designed to take the dynamic and static loads associated with the normal operation of the system.

4.3.11. Controls and Instrumentation

4.3.11.1 General System Description

1. The control and instrumentation systems shall be designed to control and monitor the various normal process functions throughout the system and will permit a safe, orderly shutdown of the system.

2. The controls and instrumentation systems will enable the operators to perform the designated functions efficiently and safely.

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

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

5. Radiation monitoring and surveillance instrumentation essential for the protection of operating personnel, the public and the environement is provided.

4.3.11.2 <u>Performance and Design Requirements</u>

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

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

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

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

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

6. Radiation monitors shall be provided for the detection of gamma radiation. In-line radiation monitors were installed to monitor beta radiation, however to date have not been used or maintained, nor are they planned to be.

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

4.4 System Operational Concepts

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

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

1. Ion-exchanger units will be preloaded with new ion exchange media prior to placement in the system. The ion exchanger units will utilize a homogeneous mixture of zeolite media.

2. Water will be introduced to fill and vent the ion-exchange units.

3. These preloaded SDS ion-exchange units will be lowered into the Unit 2 spent fuel pool and placed in the containment enclosures.

4. Inlet and outlet header connections will be made to the ion-exchange units.

5. The ion-exchange system isolation valves will be opened and treatment of the contaminated waste stream will begin at low flow rates until system integrity and acceptable out water quality are verified.

6. The flow rate to the ion-exchange units will be increased on a gradual basis until the desired operational flow rate is achieved.

7. When the first ion-exchange bed becomes depleted, the unit will be flushed with processed water to ensure that radioactive waste water in the system piping is purged prior to disconnecting the quick disconnects on the demineralizer unit.

8. The ion-exchange unit will be decoupled remotely via the use of quick disconnects and will be stored in the spent fuel pool. However, loading directly into a cask prior to shipment is possible.

9. After the first ion-exchange unit has been removed, the second ion exchange unit will be placed into the position of the first unit, and the third ion exchange unit will be moved to the second position. A new ion-exchange unit will be installed in the third position. In some instances fewer than three (3) ion-exchange units will be required to achieve the desired decontamination factors. In these cases, jumpers will be installed to bypass the unused positions.

Chapter 5

System Description and Arrangement

5.1 Demineralizer System

5.1.1 Influent Water Filtration

A flow diagram of the waste water influent system is shown in Fig. 5.1. Contaminated water is pumped into the SDS from either the containment sump or the RCS. The containment sump will employ the presently installed SWS-P-1 pump (jet pump).

Two filters have been installed to filter out solids in the untreated contaminated water before the water is processed by the ion-exchangers. These filters will be either cartridge or sand type. The cartridge filter elements are protected by 3/16 inch perforated metal plate serving as a roughing screen. The prefilter has 125 micron filter cartridges to remove debris and suspended solids from the contaminated water. The design of the final filter is simillar to the prefilter except that the filter cartridge is designed for removal of suspended solids of greater than 10 microns in size from the contaminated water. The two sand filters are loaded in layers. The first layer is 200 pounds of 0.85 mm sand and the second layer is 700 pounds of 0.45 mm sand. Borosilicate glass with a normal Boron content of 22% is added uniformly through the sand to prevent potential criticality. The flow capacity through each filter is 50 gpm. Reverse flow through filters is prevented by a check valve in the supply line to each filer.

Each filter is housed in a containment enclosure to enable leakage detection and confinement of potential leakage. The filters are submerged in the spent fuel pool for shielding considerations.

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

Inlet, outlet, and vent connections on the filters are made with quick disconnect valved couplings which are remotely operated from the top of the pool. Inlet-outlet pressure gauges are provided to monitor and control solids loading. Load limits for the filters are based on filter differential pressure, filter influent and effluent sampling, and/or the surface dose limit for the filter vessel. A flush line is attached to the filter inlet to provide a source of water for flushing the filters prior to removal.

5.1.2 Ion Exchanger Units

A flow diagram of the ion exchange manifold and primary ion-exchange columns is shown in Fig. 5.2. This system consists of six underwater columns (24 1/2 in. x 54 1/2 in.), each containing eight cubic feet of homogeneously mixed Ion Siv IE-96 and LINDE-A zeolite media and two underwater columns containing sand filter media. The six zeolite beds are divided into two trains each containing three beds (A, B, C,) with piping and valves provided to opeate either train individually or both trains in parallel.

The effluent from the first parallel train of three zeolite beds flows through either of the "cation" sand filters. Jumpers are provided to permit fewer than four (4) vessel per train operation. An in-line radiation monitor measures the activity level of the water exiting the cation exchanger. The valve manifold for controlling the operation of the primary ion exchange columns is located above the pool, inside a shielded enclosure that contains a built-in sump to collect leakage that might occur. Any such leakage is routed back to the RCS manifold. A line connects to the inlet of each primary exchanger to provide water for flushing the exchangers when they are loaded. Radionuclide loading of ion exchange vessels is determined by analyzing the influent and effluent from each exchanger. Process water flow is measured by instruments placed in the line to each ion-exchange train.

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

5.1.3 Leakage Detection and Processing

Each submerged vessel is located inside a secondary containment box that contains spent fuel pool water. During operation the secondary containment lid is closed. This lid is slotted to permit a calculated quantity of pool water to flow past the vessels and connectors. Pool water from the containment boxes is continuously monitored to detect leakage and is circulated by a pump through one of the two leakage containment ion-exchangers (See Figure 5.2). Any leakage which occurs during routine connection and disconnection of the quick-disconnects will be captured by the containment boxes, diluted by pool water, and treated by ion-exchange before being returned to the pool.

5.1.4 EPICOR-II

EPICOR-II (Figure 5.3) can provide final treatment of water after the water is processed through the SDS. When processing containment sump water, the processing plan is to polish with EPICOR-II. When processing RCS water, EPICOR II may be used as necessary to remove Antimony 125 before being returned to RCS (prior chemical adjustment will be required). EPICOR-II consists of filters, ion-exchangers and receiver tanks. The purpose of EPICOR-II is to remove trace fission products thay may be present in the water. The EPICOR-II safety assessment is provided in NUREG-0591.

5.1.5 Monitoring Tank System

Effluent from the SDS ion-exchanger can flow into one of two monitoring tanks (Figure 5.4) or in the case of RCS processing, directly to one of three RCBT's. The purpose of the monitoring tank system is to collect treated water. Each monitor tank is equipped with a sparger and tank level indicators that will automatically shut the inlet to the tank should a high level condition exist. Water in the monitoring tanks can be transferred back for reporcessing by SDS or used as flush water in the SDS, or directed to existing tankage.

5.1.6 Off-Gas and Liquid Separation System

An off-gas and liquid separation system collects gaseous and liquid wastes resulting from the operation of the water treatment system. The off-gas system is illustrated in Figure 5.5. Gaseous effluent lines from the ion exchange vessels, sampling glove boxes and shielded valving manifolds are connected to the off-gas system. Gaseous effluent is passed through a mist eliminator in the off-gas separator tank before being treated by an electric off-gas heater to reduce the off-gas relative humidity to 70%. A roughing filter and two HEPA filters are provided for further treatment. Air is moved through the system by a centrifugal blower rated at 1000 cfm. The discharge of this blower will be monitored and routed to the existing Fuel Handling Building HVAC system. Moisture collected by the off-gas system and waste returned from the continuous radiation monitoring system is directed into a separator tank. At the top of the tank a mist eliminator separates moisture from effluent gas prior to the gas entering the off-gas treatment system. The tank is located in the surge pit and is covered with a concrete and lead shield. The level in the tank will be indicated and controlled manually to return collected water to the RCS manifold for reprocessing. Offgassing of the RCBT's during processing of the RCS to the RCBT's is handled by established station procedures involving the Waste Gas Decay Tanks. Discharge from these tanks is filtered through HEPA filters before being released through the station vent.

5.2 Sampling and Process Radiation Monitoring System

The sampling glove boxes are shielded enclosures which allow water samples to be taken for analysis of radionuclides and other contaminants. The piping entering the glove boxes contains cylinders that permit draining a predetermined amount of sample into a collection bottle. Cylinders are purged by positioning valves to permit the water to flow through them and return to a waste drain header and into the off-gas separator tank. A water line connects to the inlet of the sample cylinders to allow the line to be flushed after a sample has been taken.

5.2.1 Sampling System

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

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

5.2.2 Process Radiation Monitoring System

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

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

Prior to system operation, ion exchanger and filter vessels are placed inside the containment boxes and connected with quick-disconnect couplings. When it is determined that a vessel is loaded with radioactive contaminants to predetermined limits as specified in the Process Control Program, the system will be flushed with low-activity processed water. This procedure flushes away waterborne radioactivity, thus minimizing the potential for loss of contaminants into the pool water while decoupling vessels. Vessel decoupling is accomplished remotely. Vessels are transferred using the existing fuel handling crane utilizing a yoke attached to a long shaft. The purpose of this yoke-arm assembly is to prevent inadvertent lifting of the ion exchange bed or filter vessel to a height greater than eight feet below the surface of the water in the pool. This device is a safety tool that will mechanically prevent lifting a loaded vessel out of the water shielding and preclude the possibility of accidental exposure of operating personnel.

The ion-exchange vessels are arranged to provide series processing through each of the beds: the influent waste water is treated by the bed in position "A", then by the bed in position "B", then by the bed in position "C" and finally either of the "cation" sand filters "A" or "B". The first vessel in each train (position A) will load with radioactive contaminants first. The loaded vessel will then be stored until transfer to a shielded cask. At no time during the operation of the system will a loaded vessel be taken out of the pool before it has been placed in a shielded cask. The loaded cask will be transferred from the pool with the overhead crane.

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

Figure 5.6 illustrates the arrangement of the SDS in the fuel storage pool (viewed from above). The filters, and zeolite ion exchanger vessels, are located underwater in containment enclosures in the "B" spent fuel pool. These enclosures and the exchangers are supported along one side of the pool on a structural steel rack that is attached to the pool curb. The racks act as a support for the system and also provides an operating platform from which the remote connections can be made. The off-gas system is mounted on the curb near the surge tank area.

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

5.5 Liner Recombiner and Vacuum Outgassing System (LRVOS)

The Liner Recombiners and Vacuum Outgassing System (LRVOS) is designed to eliminate the potential of a combustible Hydrogen and Oxygen mixture existing in the SDS Liners. This will facilitate the ultimate shipment and burial of the SDS Liners.

The LRVOS will perform the following operations while maintaining the normal operating depth of water between the operators and the SDS liner.

1. Reduce water in the SDS liner using vacuum outgassing to ensure enhanced operation of the recombiner catalyst.

2. Allow sampling of the liner gas at atmospheric pressures.

3. Provide capability to inert the SDS Liner with Argon or N_2 to approximately 10 psig prior to tool removal. This will prevent any water instrusion during tool decoupling.

4. Provide a means to remotely insert the recombiner catalyst into the SDS liner vent port. The catalyst is retained inside the liner by the internal vent port screen.

5. Provide sufficient recombiner catalyst to recombine the hydrogen and oxygen produced by radiolosis of the water remaining in the liner.

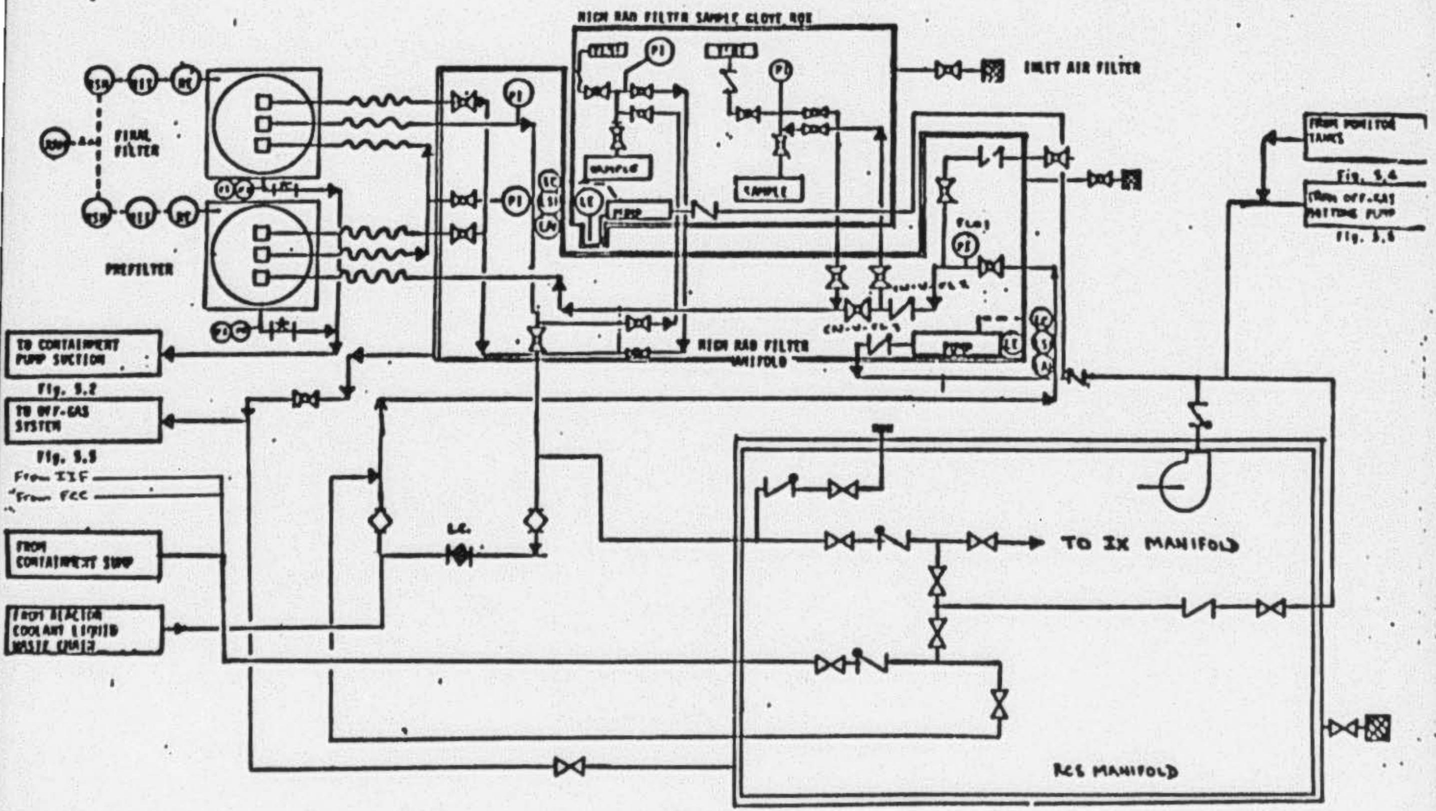

HIGH RAD FILTER SAMPLE GLOVE BOX
INLET AIR FILTER
FINAL FILTER
PREFILTER
SAMPLE
SAMPLE
PUMP
PUMP
HIGH RAD FILTER MANIFOLD
FROM MONITOR TANKS
Fig. 5.4
FROM OFF-GAS PUMP
Fig. 5.5
TO CONTAINMENT PUMP SUCTION
Fig. 5.2
TO OFF-GAS SYSTEM
Fig. 5.5
From IIF
From FCC
FROM CONTAINMENT SUMP
FROM REACTOR COOLANT LIQUID WASTE CHAIN
TO IX MANIFOLD
RCS MANIFOLD

FIGURE 5.3

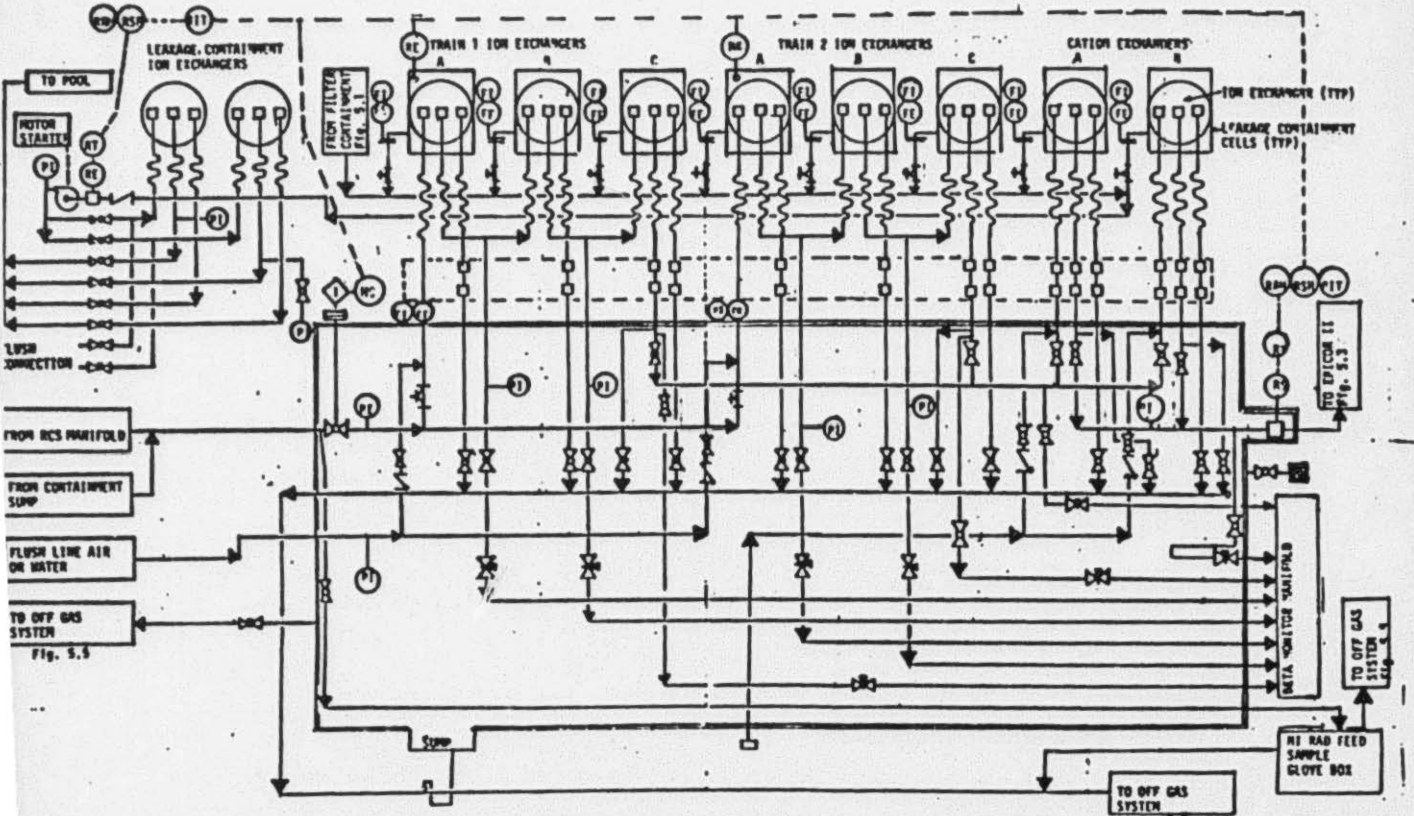


FIGURE 5.2
ION EXCHANGER TRAINS.

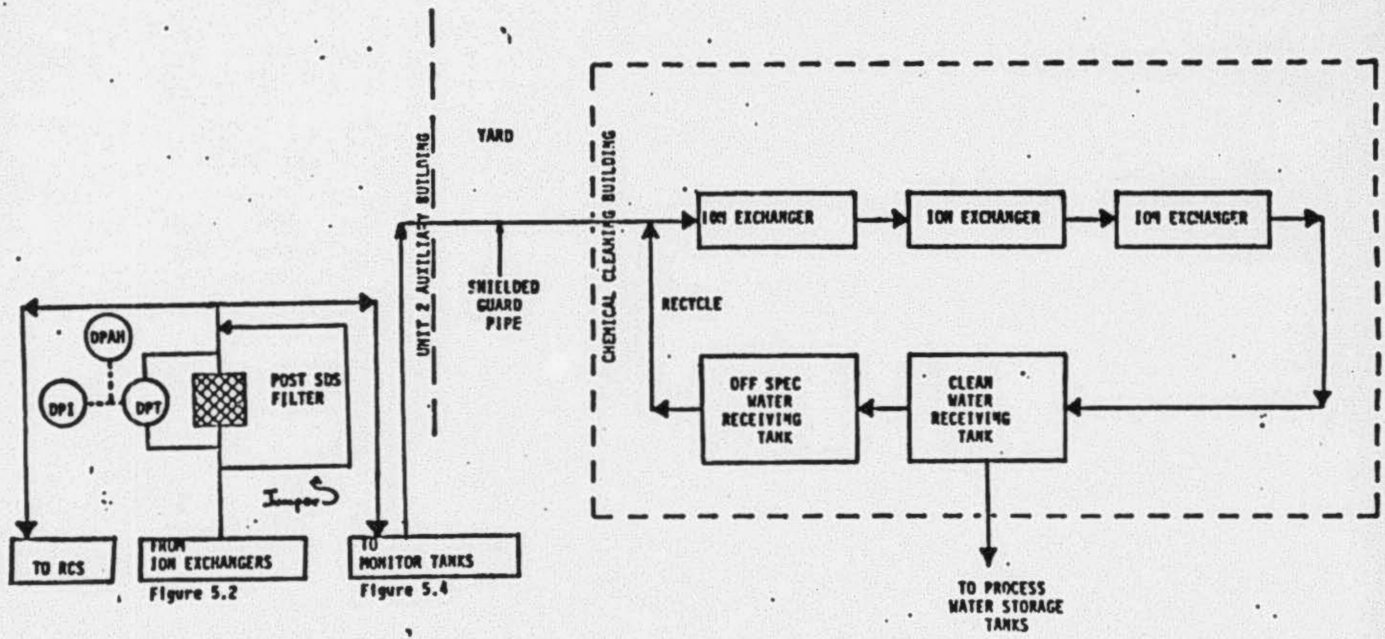
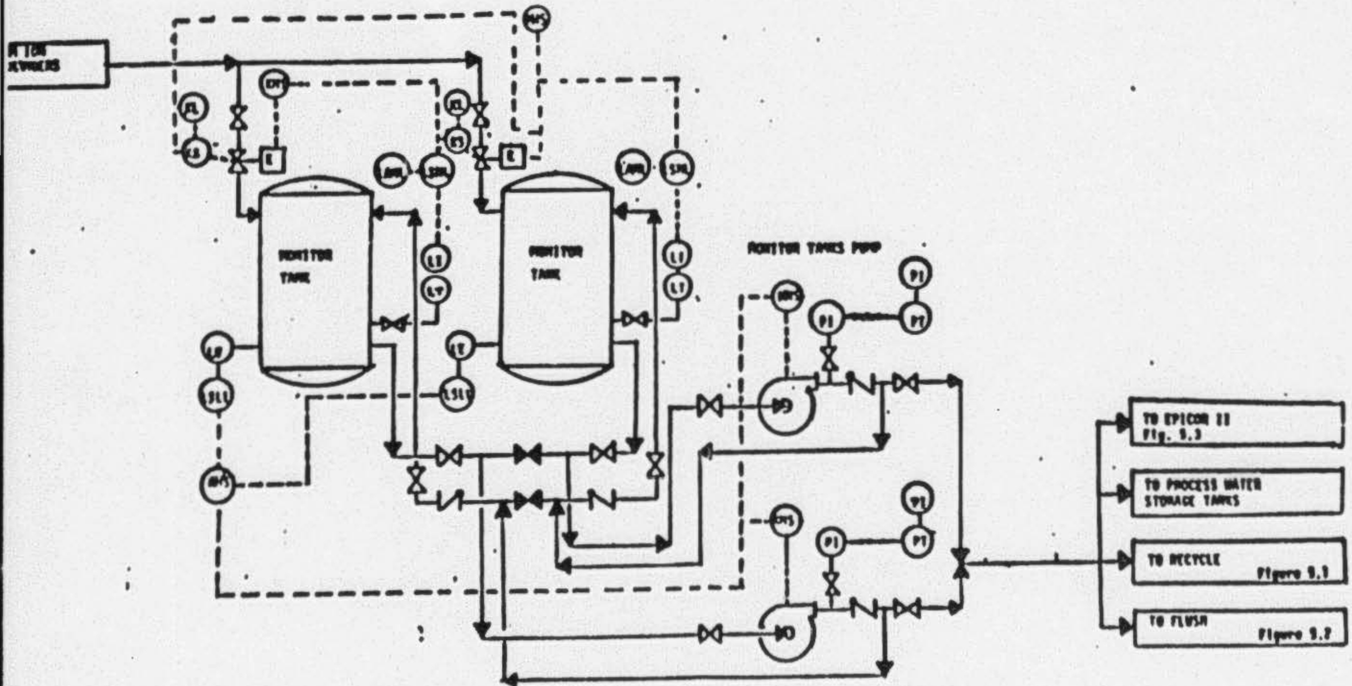


Figure 5.3 Liquid Flow Path of EPICOR II Processing System


MONITOR TANK
MONITOR TANK
MONITOR TANKS PUMP
TO EPICOR II
Fig. 9.3
TO PROCESS WATER
STORAGE TANKS
TO RECYCLE
Figure 9.3
TO FLUSH
Figure 9.2

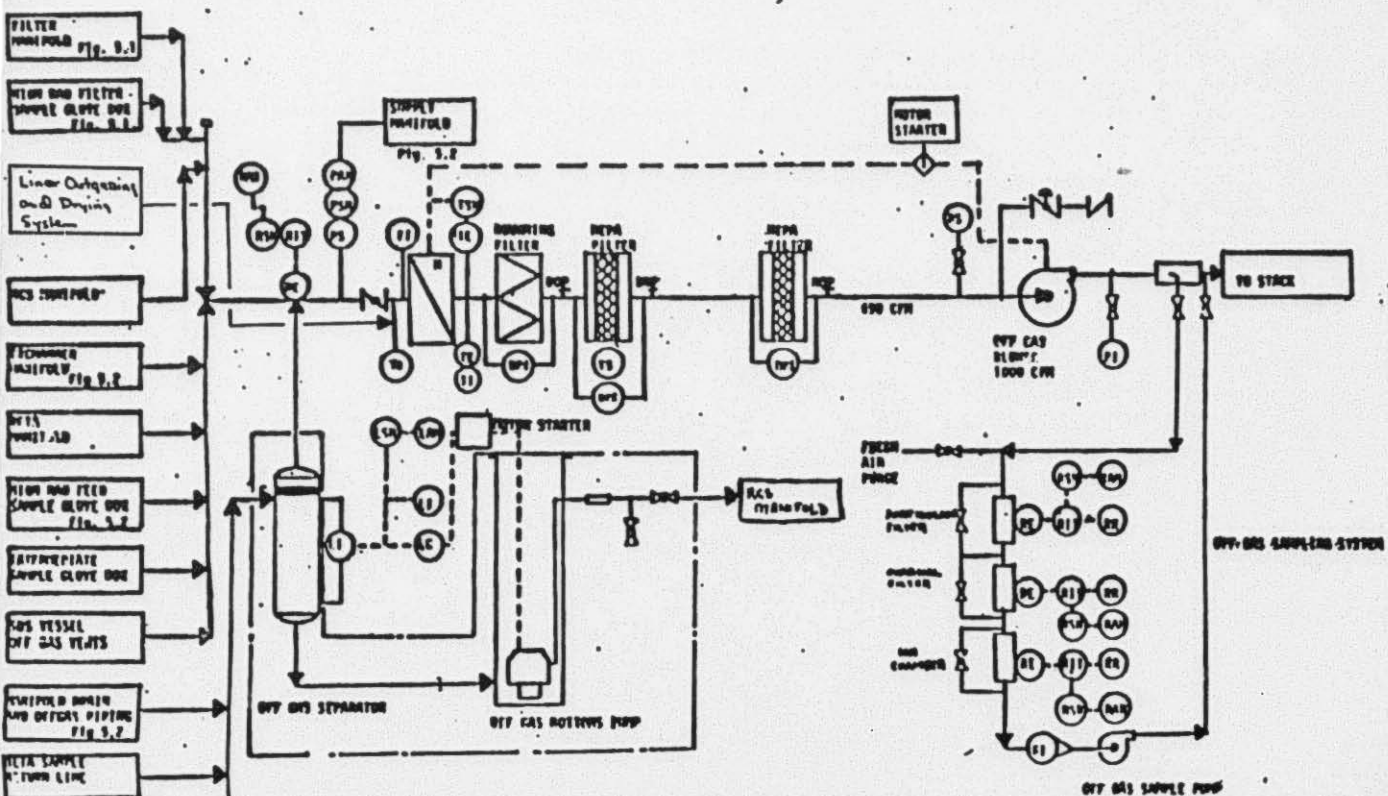
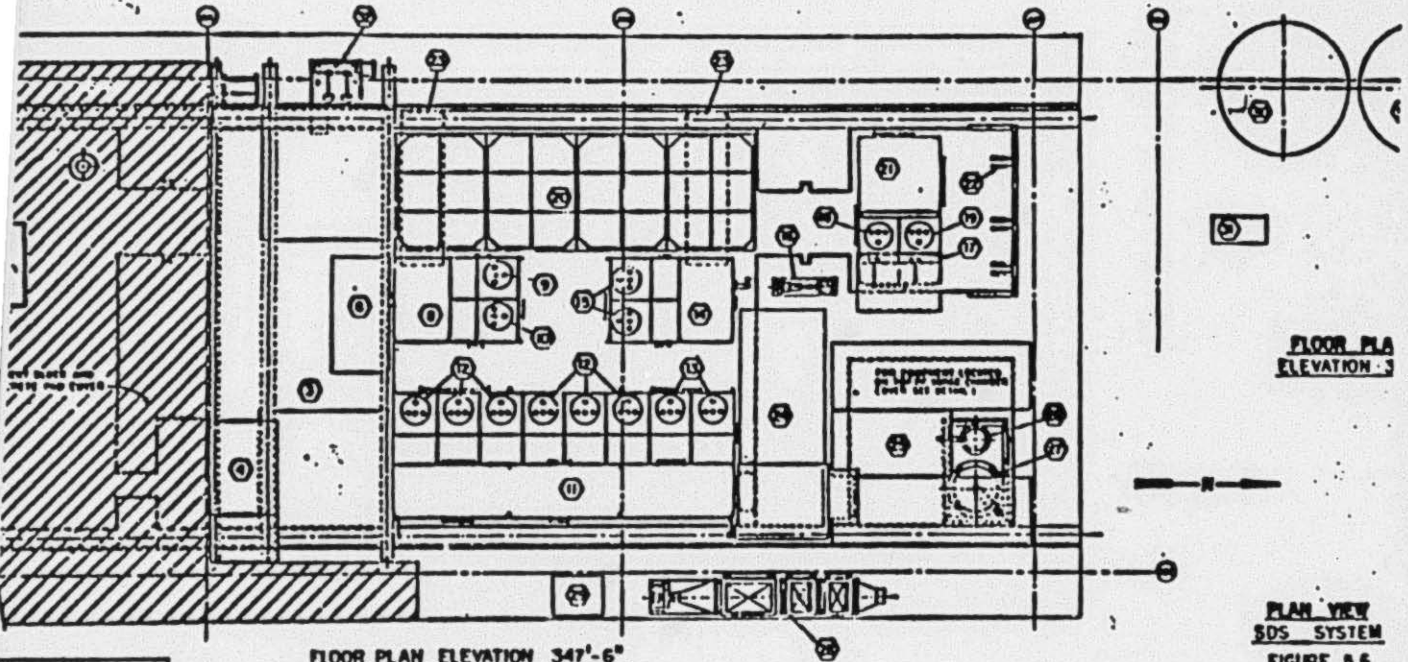
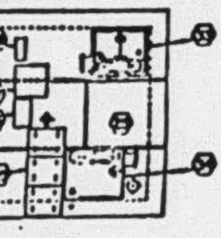


FIGURE 9.3
OFF GAS TREATMENT SYSTEM



FLOOR PLAN ELEVATION 347'-6"



DETAIL 1

PLAN VIEW
SDS SYSTEM

FIGURE 5.6

1	
2	
3	CASK SUPPORT PLATFORM
4	RCS CLEAN-UP MANIFOLD
5	
6	FILTER MANIFOLD
7	
8	FILTER SUPPORT RACK
9	PRE-FILTER
10	FINAL FILTER
11	EXCHANGER SUPPORT RACK
12	ION EXCHANGER
13	ION EXCHANGER

17	DEWATERING STATION SUPPORT RACK
18	ION EXCHANGER DEWATERING STATION
19	FILTER DEWATERING STATION
20	SPENT EXCHANGER STORAGE RACK
21	SHIPPING CASK SUPPORT PLATFORM
22	YOKE HANGER
23	MOVEABLE PERSONNEL BRIDGE
24	EXCHANGER MANIFOLD
25	SURGE CHAMBER COVER
26	OFF-GAS PUMP STAND
27	OFF-GAS SEPARATOR
28	OFF-GAS FILTER UNIT
29	OFF-GAS SAMPLING
30	MONITORING TANK

33	HI-RAD FEED SAMPLE GLOVE BOX
34	INTERMEDIATE LEVEL SAMPLE GLO
35	BETA MONITOR MANIFOLD
36	ANNUNCIATOR PANEL
37	RAD MONITOR PANEL
38	OFF-GAS SEPARATOR LEVEL INSTR.

Chapter 6

Radiation Protection

6.1 Ensuring Occupational Radiation Exposures are ALARA

6.1.1 Policy Considerations

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

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

1. Sound engineering design of the facilities and equipment.

2. The use of proper radiation protection practices, including work task planning for the proper use of the appropriate equipment by qualified personnel.

3. Strict adherence to the radiological controls procedures as developed for TMI-2.

6.1.2 Design Considerations

The SDS was specifically designed to maintain exposure to operating personnel to as low as reasonably achievable. To implement this concept the components carrying high level activity water will be provided with additional shielding or are submerged in the spent fuel pool. Shielding has been designed to limit whole body body exposure rates in operating areas to approximately 1 mR/hr. In addition, components carrying high level process fluids have been designed for exhaust to the SDS off-gas system. This method of off-gas treatment will minimize the potential for airborne releases in the work areas.

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

6.1.3 Operational Considerations

The system design reflects the following operational ALARA considerations:

1. Exposure of personnel servicing a specific component on the SDS will be reduced by providing shielding between the individual components that constitute substantial radiation sources to the receptor.

2. The exposure of personnel who operate valves on the SDS will be reduced through the use of reach rods through lead and steel shield boxes.

3. Controls for the SDS will be located in low radiation zones.

4. Airborne radioactive material concentrations will be minimized by routing the off-gas effluent from the SDS to the TMI ventilation system for further treatment.

5. The sampling stations for the feedstream and filters that contain high levels of radioactive materials will be exhausted through the SDS ventilation system.

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

6.2 Radiation Protection Design Features

6.2.1 Facility Design Features

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

1. The SDS primary demineralization units are housed under approximately 16 feet of shielding water in the TMI-2 spent fuel pool.

2. The entire process and all equipment is housed in the Auxiliary and Fuel Handling Buildings which are Seismic Category I structures with air handling and ventilation systems designed to mitigate the consequences of radiological accidents.

3. The system is designed in such a manner as to allow zero discharge of liquid effluents. The effluent processed water will be stored on the TMI site until final disposition has been determined.

4. The off-gas system effluent will be filtered and monitored before input to existing ventilation exhaust systems.

5. Filters, primary ion-exchange beds, "cation" sand filters, and their associated couplings are operated in containment devices. Each containment device is connected to a pump manifold and a continuous flow of approximately 10 GPM is maintained through each containment. The combined flow from the containment enclosures is then processed through a separate ion exchange column and then discharged back to the spent fuel pool.

6. Loaded vessels will be placed in a shielded cask underwater.

7. To the extent possible all-welded stainless steel construction is specified to minimize the potential for leakage.

8. Lead or equivalent shielding is provided for pipes, valves, and vessels (except those located under water) where necessary for personnel protection.

9. Design of a sequenced multi-bed process - three (3) beds in series to preclude breakthrough and contamination of the outlet stream.

10. The entire process stream is designed with appropriate pressure indicators.

11. Inlet, outlet and vent connection are made with remote operated-valved quick release couplings.

6.2.2 Shielding

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

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

6.2.3 Ventilation

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

1. Manual level controlled off-gas separator tank with mist eliminator to receive vent connections from the ion exchange and filter vessels, sample glove boxes, piping manifolds, and the dewatering station.

2. Roughing filter with differential pressure indication.

3. Two HEPA filters with differential pressure indication.

4. A centrifugal off-gas blower with flow indication.

5. Sample ports for monitoring the system and DOP test ports for HEPA testing.

6. The effluent of the SDS off-gas system is routed to the existing TMI-2 ventilation system exhaust, which is filtered again through the Fuel Handling Building exaust HEPA filters prior to discharge from the plant.

6.2.4 Area Radiation Monitoring Instrumentation

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

6.3 Dose Assessment

6.3.1 On-site Occupational Exposures

Normal Operation

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

A. Sampling operations
B. System start-up valve alignment
C. Spent vessel changeout
D. Cask removal, decontamination and survey operations
E. System maintenance
F. Vessel dewatering

Decommissioning

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

6.3.2 Off-site Radiological Exposures

Source Terms for Liquid Effluents

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

Radiological source terms for potential environmental releases are dependent on the processing schedule proposed for SDS and/or EPICOR-2. Review of this schedule shows that from the present (4/84) until the end of defueling, SDS, and possibly also EPICOR-2, will be dedicated to processing of RCS. Up to this time EPICOR-2 has not been used for RCS processing, but recent elevations in the Sb-125 concentration in the RCS may necessitate the use of EPICOR-2 to remove this contaminant. The assumption made here for potentil' source term generation purposes is that both SDS and EPICOR-2 will be dedicated to processing RCS. Miscellaneous small batches of liquid waste may be processed by EPICOR-2, but would be infrequent sice liners dedicated for RCS more than likely could not be used for other waste streams.

Experience with previous operations within the RCS show that minor disturbances within the reactor vessel give rise to increased concentrations of a select number of isotopes which become candidates for potential releases to systems involved in RCS decontamination and therefore, potentially to the environment. A history of concentrations of the major radiologically significant isotopes with time is shown in Figure 6-1. Not reflected in this figure are the increases in Ce-144 and alpha concentrations that accompany disturbances within the RCS. Sample analysis results, tabulated below, show typical concentrations resulting from RCS disturbances.

Radiochemistry Analysis Results

for RCS Sample of 4/9/84

(Sample #84-04966)

Isotope	Concentration (μCi/ml)	Uncertainty
Ag-110m	<1.5E-2	
Ce-144	1.1E+0	4.0E-2
Co-60	1.7E-1	1.0E-2
Cs-134	2.3E-1	1.0E-2
Cs-137	4.9E+0	4.2E-2
Ru-106	3.2E-1	5.8E-2
Sb-125	5.5E-1	3.1E-2
gross α	1.2E-3	6.1E-4
gross β	1.9E+1	2.6E-1
H-3	3.5E-2	2.2%
Sr-90	9.9E+0	35%

The increased concentration of Ce-144 and associated alpha activity is expected for RCS disturbances and is due to a colloidal suspension of finely divided fuel fines resulting from the accident. Concentration elevations of alpha bearing activity, and Ce-144, are projected to be much more significant than reflected in the table above. Short term concentration spikes may increase a factor of 10^3 or more depending on operations in the R.V.. However, for purpose of potential source term generation, these time averaged concentrations are assumed to be as tabulated above except for tritium which remains fairly stable at 0.04 μCi/ml, neglecting radioactive decay.

Source Terms for Gaseous Effluents

When the SDS Technical Evaluation Report was originally written a methodology was conceived for the definition of gaseous effluent source terms resulting from SDS/EPICOR-2 processes. This methodology used defendable, but highly conservative assumptions for defining gaseous effluent source terms. Since the beginning of SDS operation in August 1981, a significant amount of operating experience has yielded effluent data that allows more reasonable gaseous effluent source terms. The effluent data applicable to the EPICOR-2 and SDS operations is reviewed in the following section for purposes of arriving at gaseous source terms appropriate to the proposed future operations of these two systems.

A review of the 6/83 version of the SDS TER shows that, according to Table 6.2, the following quantities of the applicable isotopes would have been released to the environment over the previous 27.5 months of SDS operation through the off-gas system had the release values been correct.

Isotope	Quantity (μCi)
H-3	5.20 x 108
Sr-90	11.5μCi
I-129	4.125 μCi
Cs-134	31.6 μCi
Cs-137	280 μCi

Review of these values against airborne effluent release reports, shows the projected releases from the SDS off-gas system to be highly conservative. Because the data applicable to the SDS Off-Gas system has been reduced so that the amount attributable to this system can be separated from other sources, the following sources attributable to the future SDS/Epicor-2 operations are based on previous operations of these systems. Processed water concentrations, the ultimate source of airborne effluent concentrations, for previous operations will differ from water concentrations to be processed in the future. This initial water concentration difference has been factored into the projected release values considered for this evaluation.

SDS Off-Gas System Releases for the Period 09/15/81 to 12/31/83

SDS Off-Gas Particulate & Tritium Releases

Particulate and tritium data as measured by the Off-Gas PING-1A & H-3 bubblers was assembled for the period 9/14/81 to 12/31/83. The total amount of Tritium released through the off-gas system for this period was 7.18E-1 Curies.

The total particulates attributed to sampling through the PING-1A at the off-gas system was 3.15E-7 curies of Cs-137 and 2.52E-8 curies of Cs-134. Cs-134 appeared > LLD on one instance between 12-14-81 and 12-21-81.

The SDS Off-gas system feeds to the exhaust ventilation of the Fuel Handling Building at 1000 cfm. The point of insertion into the Fuel Handling Building exhaust is before the HEPA filters, therefore, no increase in particulate is seen at the station vent. In addition, the Fuel Handling Building exhaust is diluted by a factor of 3 by the time it reaches the station vent.

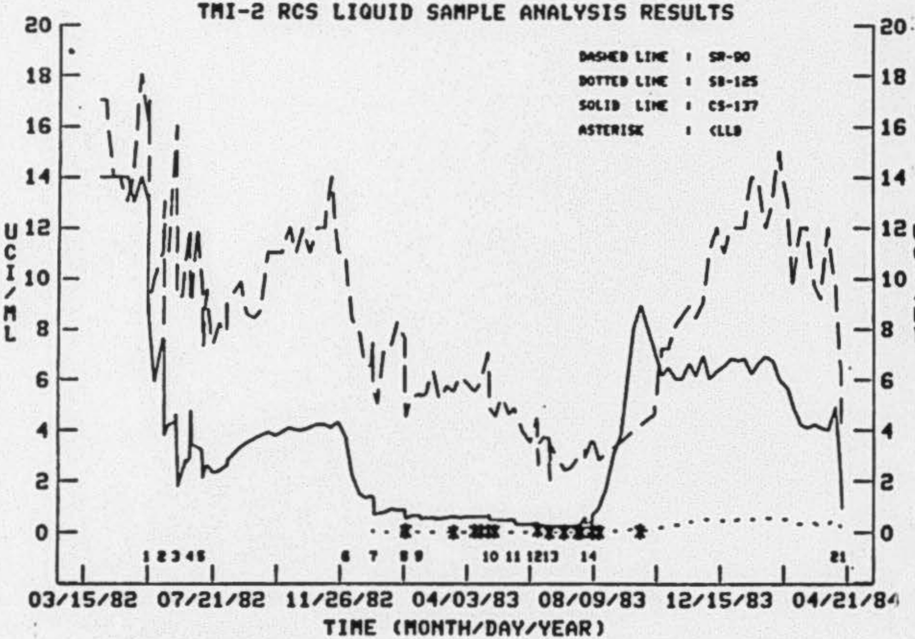
Table 6.1 lists the dates of positive particulate samples identified as Cs-137.

As a condition to startup of SDS, a tritium sampler in the off-gas system was required. A sampling unit which consists of two Fisker-Milligan bubblers in series was installed downstream of the pump of the PING-1A in the SDS off-gas system. The total cummulative curies released through the off-gas system was integrated for the time period 09/14/81 to 12/31/83 and is 7.18E-1 curies of tritium. Table 6.2 lists the H-3 curies by month and compares amounts released from the station vent, the SDS amount as a fraction of the Station Vent Release and the curies of H-3 released through EPICOR-2.

Table 6.3 shows environmental release calculations for the proposed RCS processing through SDS and EPICOR-2. The values of column 3 of the table 6.3 are about a factor of 100 lower than would have been estimated by the method of the original SER but are considered to still be conservative. The values in column 3 are the assumed values for the release rate to the environment. The values in column 4 are the concentrations at a downwind distance of 0.5 miles from the station vent, assuming atmosphere dispersion is calculated by the most restrictive data published in NUREG-0683, (Table W-3). The highest value of X/Q from this table in 3.996 E-6 sec/M^3. Using this factor and the dose conversion factor for tritium from Reg. Guide 1.109, an inhalation dose was calculated for the most restrictive recipient, an adolescent. This dose was calculated to be 1.5×10^{-5} mrem/yr.

As shown by the value of summation of the Cx/MPCx at the bottom of column 6, the total maximum yearly average concentration for all the isotopes is 16.5 million times more restrictive than allowable under the guidelines of 10 CFR 20 using the more restrictive of the "soluble"/"insoluble" form of each isotope.

DATE: April 18, 1984



EACH RCS CLEANUP BATCH 'FEED & BLEED' PERIOD IS NUMBERED ABOVE THE APPROPRIATE POINT ON THE TIME AXIS. BATCH NUMBERS REPRESENTING RCS SAMPLING BATCHES ARE NOT SHOWN.

NOTE: BETWEEN 08/01/83 AND 04/11/84, RCS WAS SAMPLED VIA CRDM H-8 ORIFICE IN THE REACTOR HEAD; BEFORE AND AFTER, VIA THE TEMPORARY NUCLEAR SAMPLING SINK.

Figure 6-1

Table 6.1

Positive Particulate Samples Identified as Cs-137

Dates	Curies of Cs-137	Curies of Cs-134
9-28-81 to 10-5-81	3.17E-9	-
12-7-81 to 12-14-81	1.64E-8	-
12-14-81 to 12-21-81	2.49E-7	2.52E-8
12-21-81 to 12-28-81	2.88E-9	-
1-18-82 to 1-25-82	4.53E-9	-
6-14-82 to 6-27-82	4.46E-9	-
9-20-82 to 9-27-82	6.16E-9	-
9-25-83 to 10-2-83	1.73E-8	-
11-20-83 to 11-27-83	1.09E-8	-
Total	3.15E-7 Curies of Cs-137;	2.52E-8 Curies of Cs-134

Table 6.2

Station Tritium Release Values

Dates	SDS Ping 1A H-3, Ci	Station Vent H-3, Ci	SDS Ping 1A H-3, fraction of Station Vent	EPICOR-II H-3, Ci
9-14-81 to 9-30-81	2.99E-2	5.24E-1	0.0367	2.91E-1
Oct. 81	5.71E-2	3.25E0	0.0176	1.03E-2
Nov. 81	1.17E-1	1.30E1	0.0090	1.20E-2
Dec. 81	6.64E-2	1.14E0	0.0582	3.10E-2
Jan. 82	5.70E-2	5.77E0	0.0099	3.06E-2
Feb. 82	2.12E-2	1.68E-1	0.1262	5.77E-3
Mar. 82	3.54E-2	3.97E1	0.0009	7.71E-1
Apr. 82	2.72E-2	1.80E0	0.0151	2.30E-3
May 82	1.02E-2	6.31E0	0.0016	1.26E-3
Jun. 82	9.80E-3	3.06E0	0.0032	6.39E-3
Jul. 82	8.50E-3	1.42E0	0.0060	6.58E-3
Aug. 82	2.17E-2	1.40E1	0.0016	1.11E-2
Sep. 82	8.80E-3	1.48E1	0.0006	1.30E-2
Oct. 82	1.38E-2	1.17E1	0.0012	1.33E-1
Nov. 82	2.84E-2	1.88E0	0.0151	6.50E-2
Dec. 82	2.05E-2	1.02E1	0.0020	2.02E-2
Jan. 83	1.44E-2	3.83E0	0.0038	3.00E-2
Feb. 83	1.08E-2	8.04E0	0.0013	1.01E-2
Mar. 83	1.05E-2	3.58E0	0.0029	6.20E-3
Apr. 83	3.00E-2	3.03E0	0.0099	1.02E-3
May 83	7.80E-3	1.61E0	0.0048	3.71E-3
Jun. 83	2.13E-2	1.33E1	0.0016	4.82E-3
Jul. 83	9.50E-3	2.13E0	0.0045	3.56E-3
Aug. 83	7.00E-3	3.15E0	0.0022	1.04E-2
Sep. 83	1.33E-3	2.60E0	0.0005	9.10E-3
Oct. 83	2.34E-2	2.15E0	0.0109	4.24E-3
Nov. 83	3.48E-2	2.41E0	0.0144	< LLD
Dec. 83	1.38E-2	2.83E0	0.0049	< LLD
Total	7.175E-1	177.4	-.----	1.44
Ci/month	2.61E-2	6.45	-.----	5.22E-2

Environmental Release Calculations for the Proposed
RCS Processing Through SDS and EPICOR-2

The amount of RCS to be processed over a years time is projected to be 1.3 x 10[6] gallons. Concentrations of the various radionuclides in this volume are assumed to be as tabulated below.

Table 6.3

RCS Processing Release Parameters

Isotope	Conc. (μCi/ml)	Ci/sec.	Conc. at 0.5 units (Ci/m5)	10 CFR 20 Table II Col. 1	Cx/MPCx
Ag-110m	<1.5E-2	<4.7E-18	<1.9E-23	3E-10	<6.3E-14
Ce-144	1.1E+0	3.4E-16	1.4E-21	2E-10	7.0E-12
Co-60	1.7E-1	5.3E-17	2.1E-22	3E-10	7.0E-13
Cs-134	2.3E-1	7.0E-17	2.8E-22	4E-10	7.0E-13
Cs-137	4.9E+0	1.5E-15	6.0E-21	5E-10	1.2E-11
Ru-106	3.2E-1	9.9E-17	3.9E-22	2E-12	2.0E-12
Sb-125	5.0E-1	1.6E-16	6.4E-22	9E-10	7.1E-13
Sr-90	9.9E+0	3.2E-15	1.3E-20	3E-11	4.3E-10
H-3	3.5E-2	2.9E-9	1.2E-14	2E-7	6.0E-8
U-235*	3.8E-7	1.2E-22	4.8E-28	4E-12	1.2E-16
U-238*	2.4E-6	7.4E-22	3.0E-27	3E-12	1.0E-15
Pu-238*	4.7E-7	1.5E-22	6.0E-28	7E-14	8.6E-15
Pu-239*	8.4E-4	2.6E-19	1.0E-24	6E-14	1.7E-11
Pu-240*	2.1E-4	6.5E-20	2.6E-25	6E-14	4.3E-12
Pu-241*	1.4E-2	4.3E-18	1.7E-23	3E-12	5.7E-12
Am-241*	1.4E-4	4.3E-20	1.7E-25	2E-13	8.5E-13
Np-237*	1.1E-7	3.4E-23	1.4E-28	1E-13	1.4E-15
Np-339*	1.7E-8	5.3E-24	2.1E-29	2E-8	1.1E-21
(Gross α)	(1.2E-3)	(3.7E-19)	(1.5E-24)	(2E-14)	-.-----

TOTAL $\frac{Cx}{MPCx}$ = 6.05E-8

* Values calculated according to the Ce-144/fuel ratio value is calculated by the ORIGEN Computer code as programmed for the TMI-2 Operational history and a decay time of 5.5 years.

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Chapter 7

Accident Analysis

Because of the inherent safety features of the Submerged Demineralizer System and maximum utilization of existing site facilities, potential accidents which involve the release of radionuclides to the environment are minimized. Hypothetical accidents during system operations are proposed and evaluated in the following assessment. The following accident analysis has been performed based on the assumption that zeolite beds are radiologically loaded to 60,000 Ci. Should higher radiological loadings be determined to be appropriate, the accident analysis will be reassessed using the higher radiological loadings.

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

Assumptions:

The effluent line from the final filter develops a leak and is not detected immediately. Contaminated water is released into the pool at a rate of 15 gpm for a period of 15 minutes, (225 gallons or ~15 curies).

It is assumed that the total activity is made up of 0.2Ci of Cs-134 and 4.2 Ci of Cs-137, 0.94 Ci of Ce-144, 8.4 Ci of Sr-90, and 0.5 Ci of Sb-125 (based upon the measured concentrations as reported in Chapter 6). Analysis of the accident also assumes uniform mixing in 233,000 gallons of pool water and results in pool water contamination levels of

0.017 uCi/ml of total activity or of 0.0075 μCi/ml of gamma emitters. This value is only about 3% of the value calculated for the same accident assuming RB "sump" water was inadvertently pumped into the fuel pool water.

Occupational Exposure Effects:

The dose rate is calculated to an individual on the walkway at a point three feet above the surface of the water using the ISOSHLD-II computer code. The depth of water in the pool is 38 feet. The calculated maximum exposure rate at three feet above the surface is 4.2 mR/hr.

After such an accidental leak the pool would contain ~1 millicurie of alpha activity. Such a leak would require that more stringent contamination control procedures would have to be installed to prevent alpha activity from leaving the pool. Cleanup of the pool would require passing the water through 2 specially prepared 4x4 liners; one similar to the SDS liners and one similar to the EPICOR.

Off-site Effects:

A review of previous SDS operation shows that this accident does not release measurable activity to the environment.

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

Conclusions:

This hypothetical accident is evaluated under conservative assumptions.

Although the analysis of this hypothetical accident provides results that indicate radiation field of 4.2 mR/hr at a level three feet above the pool surface, area radiation monitor alarms would indicate its presence. Personnel would be evacuated to ensure that occupational exposures are limited.

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

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

Assumptions:

A pipe rupture occurs in the inlet line to the filters above water level at the southeast corner of the pool. The leak proceeds for fifteen minutes before the pump is stopped. Contaminated water sprays from around the lead brick shielding. A total of 38 gallons of water is spread onto a surface area of 100 ft.^{2} and 340 gallons of contaminated water are drained into the pool. It is further assumed that the contaminated water contains 0.065 Ci/gallon of activity in the same concentration ratios that were assumed for the previous hypothetical accident.

Occupational Exposure Effects:

As a result of this hypothetical accident, five significant effects are postulated:

1. The maximum gamma exposure rate at the surface of the contaminated floor area is calculated to be 100 mRem/hr.

2. The maximum beta exposure rate at a point three feet above the surface of the contaminated floor area is estimated to be 560 mRad/hr.

3. The exposure rate from the surface of the contaminated spent fuel pool waters, at a point three feet above the surface, would be approximately 6.3 mRem/hr gamma, and ~32 mRad/hr beta.

4. The pool water would contain about 1.5 millicuries of alpha activity, and

5. the floor surface would be contaminated with about 0.2 millicuries of alpha activity.

Offsite Effects:

To calculate off-site concentrations it is conservatively assumed that 0.1% of the activity sprayed from the pipe becomes airborne within the Fuel Handling Building. This airborne activity is evacuated from the Fuel Handling Building by the FHB H&V system which is filtered through HEPA filters before the airborne effluent reaches the environment. The offsite concentration is maximized by assuming the activity is evacuated from the FHB in a 15 minute time period and, consequently, the hypothetical release to the environment occurs over a 15 minute period. Release parameters for this accident are as tabulated below. Credit has been taken for only 1 of the 2 HEPA filter banks of the FHB exhaust filter system.

Conclusions:

Analysis of this hypothetical accident, show that even under the conservative assumptions of the accident, the effluent concentrations, for a period of 15 minutes, are calculated to reach a level such that the summation of the individual. Ci/MPC, values is 79% of the allowable. Credit for the neglected HEPA filter and a less conservative X/Q would reduce this fraction to an even lower value.

Release Parameters for a RCS Pipe Spray Leak

Isotope	Release rate to FHB (ci/s)	Accident Station Vent Release Rate (ci/s)	EA Concentration (Ci/M³) (at 610m with X/Q=1.3x10⁻³ S/M³)*	Cx/MPC.
Ag-110m	<2.4E-8	<2.4E-11	<3.1E-14	<1.0E-4
Ce-144	1.8E-6	1.8E-9	2.3E-12	1.2E-2
Co-60	2.7E-7	2.7E-10	3.5E-13	1.2E-3
Cs-134	3.7E-7	3.7E-10	4.8E-13	1.2E-3
Cs-137	7.8E-6	7.8E-9	1.0E-11	2.0E-2
Ru-106	5.1E-7	5.1E-10	6.6E-13	3.3E-3
Sb-125	8.0E-7	8.0E-10	1.0E-12	1.1E-3
Sr-90	1.6E-5	1.6E-8	2.1E-11	7.0E-1
H-3	5.6E-8	5.6E-11	7.3E-11	3.7E-4
U-235	6.1E-13	6.1E-16	7.9E-19	2.0E-7
U-238	3.8E-12	3.8E-15	4.9E-18	1.6E-6
Pu-238	7.5E-13	7.5E-16	9.8E-19	1.4E-5
Pu-239	1.3E-9	1.3E-12	1.7E-15	2.8E-2
Pu-240	3.4E-10	3.4E-13	4.4E-16	7.3E-3
Pu-241	2.2E-8	2.2E-11	2.9E-14	9.7E-3
Am-241	2.2E-10	2.2E-13	2.9E-16	1.5E-3
NP-237	1.8E-13	1.8E-16	2.3E-19	2.3E-6
NP-239	2.7E-14	2.7E-17	3.5E-20	1.8E-12

$$\text{TOTAL } \frac{\text{Ci}}{\text{MPC.}} = 0.786$$

* The X/Q value chosen for this analysis (1.3×10^{-3} S/M³) was used because of the short duration of the release. This precluded the use of the annual average X/Q.

As shown at the bottom of column 5, the summation of the $\frac{\text{Cx}}{\text{MPC.}}$ is only 79% of the specified 1.0 for this scenario.

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

7.3 Inadvertent lifting of prefilter above pool surface

Assumptions:

It is assumed that due to a failure in the crane control system, the over head crane moves toward the loading bay after pulling one expended filter to the maximum height of eight feet below the pool surface. As the crane moves toward the bay, the handling tool hits the end of the pool and the filter is dragged from the water exposing operating personnel.

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

Occupational Exposure Effects:

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

Off-site Effects:

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

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

Conclusions:

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

7.4 Inadvertent lifting of zeolite ion exchanger above pool surface

Assumptions:

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

Analysis of the accident is performed by modeling the zeolite ion exchanger bed in cylindrical geometry and calculating the dose rate at a distance of 20 feet from the surface of the zeolite ion exchanger. The calculated dose rate is approximately 340 Rem/hr based on an estimated zeolite ion exchange bed loading of approximately 2730 Curies of Cesium-134 and approximately 51,900 Curies of Cesium 137.

Occupational Exposure Effects:

As the zeolite vessel nears the surface of the spent fuel pool water, area radiation monitor alarms will automatically sound announcing the presence of high radiation fields. Personnel would be evacuated from the area to reduce occupational doses. Airborne contamination would not occur since the activity is fixed on the zeolites.

Offsite Effects:

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

Conclusions:

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

7.5 Inadvertent Drop of SDS Shipping Cask

Assumptions:

It is assumed that due to a failure in SDS shipping cask handling equipment an SDS cask containing a zeolite ion exchanger is dropped from the Fuel Handling Building (FHB) crane to the floor at EL 305'. The SDS shipping cask is assumed to drop from the maximum crane lift height. Upon impact with the floor at EL 305', the SDS shipping cask is assumed to experience rupture as well as rupture of the zeolite vessel, thus exposing the dewatered zeolite resins to the FHB atmosphere. The radiation source is approximately 2730 Curies of Cs-134 and approximately 51,900 Curies of Cs-137 on the zeolite ion exchange media. The contribution from other isotopes on the zeolite media and residual containment building sump water (Table 1.1) in the ion exchange media is negligible; it is assumed that a factor of 10^{-4} of the isotopes are instantaneoulsy released to the FHB atmosphere. This assumption is conservative because the isotopes are absorbed onto the zeolite media. The Fuel Handling Building HEPA filters are assumed to have an efficiency of 99%.

Occupational Effects:

Assuming that the SDS shipping cask ruptures completely exposing the zeolite ion exchanger containing the activity mentioned above, the calculated dose rate is approximately 340 Rem/hr at a distance of 20 feet. Upon the rupture of the cask, radiation monitors will sound announcing the presence of high radiation fields. Personnel would be evacuated from the area to reduce radiation exposures. Airborne contamination will not occur if the zeolite ion exchange vessels remains intact. With the assumption that the vessels rupture and radioactive material becomes airborne, the airborne activity will be reduced to acceptable levels by the Fuel Handling Building HVAC System prior to atmospheric release.

Operational Effects:

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

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

Off-Site Effects:

The increase in radiation level at the site boundary would not be significant due to the shielding provided by the FHB walls and the distance to the site boundary, if the SDS cask ruptures exposing the zeolite ion exchanger. With the assumption that radioactive material escapes, the whole body dose due to the released activity at the site boundary will be less than 1 mrem for both beta and gamma radiation.

Conclusions:

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

Campbell, D.O., E.D. Collins, L.J. King, J.B. Knauer, "Evaluation of the Submerged Demineralizer System (SDS) Flowsheet for Decontamination of High-Activity-Level Water at the Three Mile Island Unit 2 Nuclear Power Station," ORNL/TM-744:, July 1980.

Clark, W.E., "The Use of Ion-Exchange to Treat Radioactive Liquids in Light-Water-Reactor Nuclear Reactor Power Plants," NUREG/CR-0143, ORNL/NUREG/TM-204 (August 1978).

Ga, J. H., E. W. Murbach, and A. K. Williams, 1979, "Experience and Plans for Effluent Control at LWR Fuel Reprocessing Plants", in Proc. Conf. on Controlling Airborne Effluents from Fuel Cycle Plants, AICHE Topical Meeting.

Lin, K. H., "Use of Ion-Exchange for the Treatment of Liquids in Nuclear Power Plants," ORNL-4792 (December 1973).

Lin, K. H., "Performance of Ion-Exchange Systems in Nuclear Power Plants," AICHE Symposium, Ser. 71 (152), pp 224-35 (1975).

Willingham, W. E., 1972, "The Vitro Engineering ISOSHLD User's Manual," VITRO-R-153.

U.S. Department of Health, Education, and Welfare, 1970, Radiological Health Handbook, U.S. Government Printing Office, Washington, D.C.

Chapter 8

Conduct of Operations

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

8.1 System Development

System development activities have been performed to assure that components are developed specifically to meet the conditions imposed at TMI and perform in the intended manner.

The ion-exchange process is a well understood process. Even though ion-exchange media have been in use for approximately 50 years or more, a development program was conducted at the Oak Ridge National Laboratory, the results of which are documented in ORNL TM-7448, to ensure that the media selected for use at TMI provided optimized performance characteristics of various media using samples of the waters to be processed at TMI. In some cases, SDS effluent will be polished by EPICOR-II.

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

8.2 System Preoperational Testing

Prior to use in the SDS each vessel will be hydrostatically tested in conformance with the requirements of applicable portions of the ASME Boiler and Pressure Vessel Code. Upon completion of construction, the entire system will be pneumatically tested to assure leak-free operations. The system will be tested to an internal pressure of no less than 1.5 times the design pressure.

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

8.3 System Operations

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

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

8.4 System Decommissioning

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

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

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

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

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

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

Appendix No. 1

to

Submerged Demineralizer System

Technical Evaluation Report

REACTOR COOLANT PROCESSING PLAN

WITH THE REACTOR COOLANT SYSTEM

IN A PARTIALLY DRAINED CONDITION

CONTENTS

CONTENTS (continued)

Chapter 1

SUMMARY OF TREATMENT PLAN

1.1 Project Scope

The decontamination of the TMI-2 Reactor Coolant System (RCS) requires the processing of the radioactive contaminated water to reduce the activity therein. The present activity level of this water is given in Table 1.1. To date, in excess of 700,000 gallons of water have been processed from the RCS. The feed and bleed operation via the Submerged Demineralizer System (SDS) has reduced the radionuclide concentration of the RCS water: specifically the Cs-137 concentration has been reduced from 14.0 μCi/cc to the present value of approximately 0.3 μCi/cc.

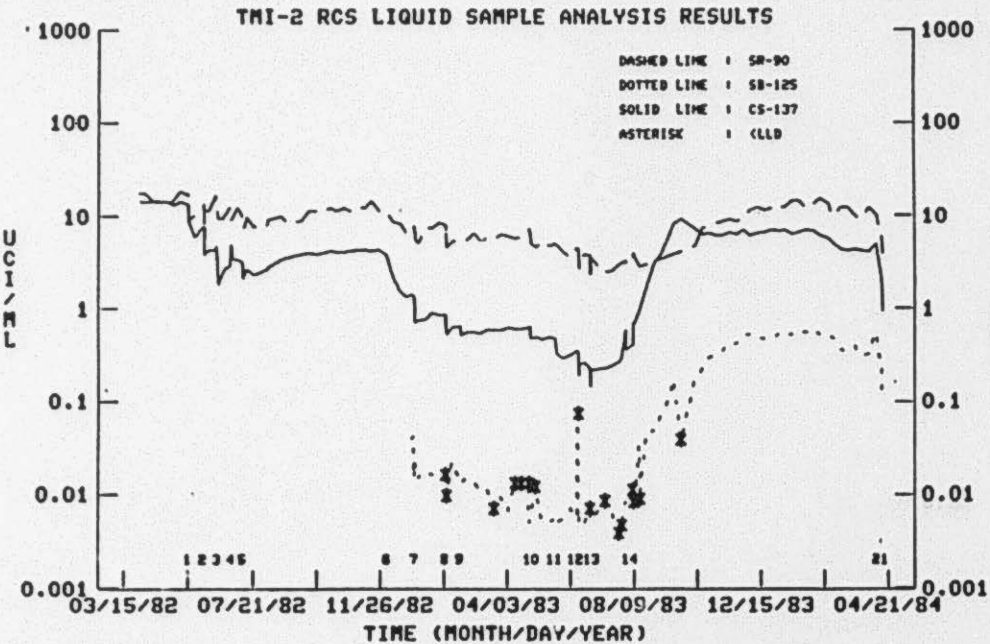
This report describes the processing of the RCS by the SDS while maintaining the RCS in the partially drained, open condition. The design features of this processing method will utilize:

1. proven processing capabilities of the SDS, and
2. Existing plant systems in support of the SDS.

1.2 Current RCS Radionuclide Inventory and Chemistry

Water samples have been taken continuously from the RCS to identify specific radionuclides and concentrations, and plant chemistry. Typical results are listed in Table 1.1. This data is based on actual samples taken. RCS activity is decreasing due to radioactive decay and leakage from the RCS which is being made up by injection of clean water into the RCS, and due to batches which have been removed for SDS processing. Figure 1.1 shows how activity for the major nuclides has decreased with respect to time. Currently Sb-125 concentrations have risen to radiologically significant levels due to changing RCS chemistry parameters. Evaluations are currently under way to determine the impact of this increased Sb-125 concentration and to determine the optimum method for its removal.

DATE: April 18, 1984



EACH RCS CLEANUP BATCH 'FEED & BLEED' PERIOD IS NUMBERED ABOVE THE APPROPRIATE POINT ON THE TIME AXIS. BATCH NUMBERS REPRESENTING RCS SAMPLING BATCHES ARE NOT SHOWN.

NOTE: BETWEEN 08/01/83 AND 04/11/84, RCS WAS SAMPLED VIA CRDM H-8 ORIFICE IN THE REACTOR HEAD; BEFORE AND AFTER, VIA THE TEMPORARY NUCLEAR SAMPLING SINK.

Figure 1.1

X2980

1.3 RCS Processing Description

On a batch basis, radioactive RCS water is letdown to Reactor Coolant Bleed Tank (RCBT) "C" while clean water is injected into the RCS from RCBT "A". RCS water is then pumped from RCBT "C" through the prefilter and final filter. RCS water then goes through the RCS manifold and the SDS ion exchangers. The effluent from the ion exchangers is routed through the cation sand filter to RCBT "A" for chemical adjustment, if necessary, and injection back into the RCS as makeup. The above process is repeated until the RCS water is decontaminated. EPICOR II may be used for processing selected batches of RCS water unless needed for chloride control.

The processing of the RCS will use the existing filter and ion exchangers of the SDS. Existing sampling connections will be used on the influent and effluent of the filters and ion exchangers to determine radionuclide and chemical composition of the RCS before and after processing.

As described in the SDS TER, the prefilters, final filters, and cation sand filters for the removal of particulate matter. The prefilter and final filter are followed by a series of ion exchange vessels containing about 8 cubic feet of zeolite ion exchange media. Location, operation, and handling of these vessels remains unchanged from the mode of operation used for processing of the Reactor Building sump water and the RCS water as described in the SDS TER.

TABLE 1.1

RCS RADIONUCLIDE AND CHEMISTRY DATA
(June 15, 1984)

ISOTOPE	RADIONUCLIDE CONCENTRATION uCi/cc
H-3	0.045
Sr-90	2.7
Cs-134	0.015
Cs-137	0.3
Gross Beta	4.1
pH	7.72
Boron	5200 ppm
Na	1500 ppm
Cl	1.25 ppm

Chapter 2

RCS PROCESSING PLAN DESIGN CRITERIA

2.1 Introduction

This RCS Processing Plan is designed to use the Submerged Demineralizer System (SDS) and portions of existing plant liquid radwaste disposal systems to decontaminate the RCS water. This will reduce plant personnel and off site radiation exposures. The design objectives of this processing plan are to utilize:

1. A system that is as independent as possible from existing radioactive waste systems at TMI-2. The SDS portion of this plan is a temporary system for the recovery of TMI-2. Only small sections of existing TMI-2 plant systems will be used.
2. A system that has proven performance in processing radioactive waste. The SDS portion of this processing plan has successfully decontaminated the Reactor Building sump and the RCS water.

2.2 Design Basis

2.2.1 Submerged Demineralizer System

The Submerged Demineralizer System was designed in accordance with the following regulatory documents:

1. Code of Federal Regulations, 10CFR20, Standard for Protection against Radiation.
2. Code of Federal Regulations, 10CFR50, Licensing of Production and Utilization Facilities.
3. U.S. Regulatory Guide 1.21, dated June 1974.
4. U.S. Regulatory Guide 1.140, dated March 1978.
5. U.S. Regulatory Guide 1.143, dated July 1978.
6. U.S. Regulatory Guide 8.8, dated June 1978.
7. U.S. Regulatory Guide 8.10, dated May 1977.

The design basis for the SDS is presented in greater detail in Chapter 4 of the SDS Technical Evaluation Report.

2.2.2 Interfacing Systems

The interfacing systems with the SDS in the RCS Processing system are:

1. Radwaste Disposal (Reactor Coolant Liquid) System
2. Reactor Coolant Makeup and Purification System
3. Auxiliary and Fuel Handling Buildings Heating Ventilation and Air Conditioning Systems
4. Nitrogen Supply System
5. Decay Heat Removal System
6. Waste Gas System
7. Standby Pressure Control System
8. Spent Fuel Cooling System
9. Instrument Air System

The design criteria for these systems (except SPC) are presented in Chapter 3 of the TMI-2 FSAR. Conformance to these criteria is presented in the respective sections for these systems in the TMI-2 FSAR. Standby Pressure Control System data may be found in section 1 of the TMI Recovery System Descriptions and TER's.

2.3 RCS Processing Plan Goal

The goal of the RCS Processing Plan is to reduce the total radionuclide concentration of Cs in the RCS to less than 1 uCi/cc. The RCS Chemistry will be maintained as follows as a minimum:

Chlorides	< 5 ppm
Oxygen	Atmospheric concentration
pH	> 7.5 but < 8.4
Boron	> 4950 ppm

The processing of water through the SDS is not expected to have any undesirable effect on the chemical characteristics of the RCS water. Maintaining proper chemistry of the makeup water will ensure that there will be no adverse effects on the RCS with respect to corrosion. The boron concentration of the makeup will also ensure that sufficient boron is present to maintain the core in a non-critical safe condition. Sampling of the RCS water will be continued by taking samples via the CRDM nozzle in accordance with approved operating procedure.

Chapter 3

SYSTEM DESCRIPTION AND OPERATIONS

3.1 Introduction

This RCS Processing Plan is designed specifically for the controlled decontamination of the radioactive water in the RCS and the treatment of the radioactive gases and solid radioactive waste which are produced. This plan will use the SDS as the means of decontamination of the RCS with support from other existing plant systems.

3.1.1 Submerged Demineralizer System

The SDS consists of a liquid waste processing system, an off gas system, a monitoring and sampling system, and solid waste handling system. The liquid waste processing system decontaminates the RCS water by a process of filtration and demineralization. The off gas system collects, filters, and absorbs radioactive gases produced during processing, sampling, dewatering, and spent SDS liner venting. The sampling system provides measurement of process performance. The solid waste handling system is provided for moving, dewatering, storage, and loading of filters and demineralizer vessels into the shipping cask. The SDS will be unchanged from that described in the SDS TER.

3.1.2 Interfacing Systems

Interfacing with the SDS are existing plant systems, as given in Section 2.2. The Reactor Coolant Liquid Waste Chain provides a staging location for the SDS for collecting and injection of RCS water from and to the RCS. The Fuel Handling Building and Auxiliary Building HVAC systems provide tempered ventilating air and controlled air movement to prevent spread of airborne contamination with the plant and to the outside environment. The Nitrogen Supply system provides N_2 for blanketing the Reactor Coolant Bleed Tanks. The Makeup and Purification and Spent Fuel Cooling Systems provide piping for the transfer of the waste water. The Waste Gas System processes the gases from the vents from the RCBT's. The Instrument Air System provides air pressure for air-operated valves in the Interfacing Sytems. The Standby Pressure Control System, installed as a temporary TMI-2 recovery system, will be used as a backup system to ensure a source of additional makeup to the RCS.

3.2 RCS Water Processing Preparation

3.2.1 RCS Preparation

The RCS will be maintained in a partially drained condition vented to atmosphere. Its water level may vary from Elevation 347' to 323'6" depending on the needs for access to the reactor vessel.

The minimum water level is expected to be 323'6" (1' above the reactor vessel flange).

At this level and at all levels above this, the Waste Transfer pumps will be used to inject RCS makeup water into the RCS for the RCS cleanup process. The maximum discharge pressure of these pumps is 74 psig at a flow rate of 40 gpm. Flow to the RCS will be controlled by valve WDL-V-36A or 36B depending on which waste transfer pump is used for feed and, if necessary, MU-V-9. MU-V10 will also be open to permit makeup flow to the RCS. The flow rate to the RCS will be maintained at less than 5 gpm to match the letdown flow rate. Minor adustments in flow rate will be made to maintain the RCS water level within the limits required.

The decay heat analysis as reported in Appendix B TMI-2 Decay Heat Removal Analysis, April 1982, submitted as a part of the Safety Evalution for Insertion of a Camera into the Reactor Vessel Through a Leadscrew Opening Rev. 2 July 1982, is applicable for the RCS processing described herein. The average incore coolant temperature will be limited to less than 170°F. This criterion was adopted as a conservative value for the recovery program to maintain a positive margin to boiling.

3.2.2 SPC Operation

The Standby Pressure Control System (SPC) will serve as a backup system to ensure that the RCS level is maintained during RCS processing.

3.2.3 Reactor Coolant Liquid Waste Chain

Prior to starting RCS water processing, the RCBT "A" will be filled with more than 50,000 gallons of borated, suitable, processed water. The radionuclide and chemistry data for this water will be similar to that used for RCS makeup during the previous RCS processing period. Chemicals will be added to this water if required to ensure that this water complies with the plant chemistry specified in Section 2.3.

3.3 RCS Water Letdown and Injection

RCS letdown will be performed by a bleed and feed process of simultaneously removing the radioactive RCS water and injecting borated processed water at the same flow rate to maintain RCS water volume constant. The bleed and feed process will be controlled from the Control Room in coordination with the Radwaste Control Panel. The RCS water is letdown through the normal letdown line on the loop cold leg before Reactor Coolant Pump RC-P-1A. The letdown rate is 5 gallons per minute if the waste transfer pumps are used or 10 gpm if a newly installed sandpiper pump (fig. 3.4), which is normally disconnected, is used. The RCS water is letdown through the letdown coolers to RCBT "C". The plugged block orifice and isolated Makeup Demineralizers and Filters are bypassed. As the RCS water is letdown, simultaneously the borated processed water located in RCBT "A" is injected to the RCS. After RCBT "C" has been filled to more than 50,000 gallons, the letdown and injection of water from and to the RCS will be secured. RCBT "C" will be recirculated prior to processing. After recirculating, decontamination of the RCS radioactive water by the SDS will commence.

3.4 RCS Processing By SDS

3.4.1 RCS Water Filtration

Two filters have been installed to filter out solids in the untreated contaminated water before the water is processed by the ion exchangers. Both filters are sand type. The two sand filters are loaded in layers. The first layer is 0.85 mm sand and the second layer is 0.45 mm sand. Mixed uniformly with the sand is approximately 6 pounds borosilicate glass which is at least 22 weight percent boron. The loading of these filters may be changed if applicable. The purpose of the borosilicate is to prevent the possibility of criticality should any fuel fines be transported in the let down. The flow capacity through each filter is 50 gpm. Reverse flow through filters is prevented by a check valve in the supply line to each filter.

Each filter is housed in a containment enclosure to enable leakage detection and confinement of potential leakage. The filters are submerged in the spent fuel for shielding considerations. Contaminated water can be pumped through the filters and the RCS manifold to the ion exchangers.

3.4.1 RCS Water Filtration (cont'd)

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

Inlet, outlet, and vent connections on the filters are made with quick disconnect valved couplings which are remotely operated from the top of the pool. Inlet/outlet pressure gauges are provided to monitor and control solids loading. Load limits for the filters are based on filter differential pressure, filter influent and effluent sampling, and/or the surface dose limit for the filter vessel. A flush line is attached to the filter inlet to provide a source of water for flushing the filters prior to removal.

3.4.2 RCS Water Demineralization

This system consists of eight underwater columns (24 1/2" x 54 1/2"), each capable of containing eight cubic feet inorganic zeolite sorbent. Homogeneously mixed Ion Siv IE-96 and LINDE-A zeolite are the medias of choice to efficiently immobilize the Cesium and Strontium in the RCS. Six zeolite beds are divided into two trains each containing two or three beds (A, B, C) with piping and valves provided to operate either train individually or both trains in parallel.

3.4.2 RCS Water Demineralization (cont'd)

The effluent from the zeolite trains flows through the remaining "cation" sand vessel. Jumpers are provided to permit 2, 3, or 4 vessels per train operation. An in-line radiation monitor measures the activity level of the water exiting the last ion exchanger vessel. The valve manifold for controlling the operation of the primary ion exchange columns is located above the pool, inside a shielded enclosure that contains a built-in sump to collect leakage that might occur. Any such leakage is routed to the off gas bottoms separator tank and pump. A line connects to the inlet of each ion exchanger to provide water for flushing the ion exchangers when they are loaded. Radionuclide loading of ion exchange vessels is determined by analyzing the influent and effluent from each exchanger.

Process water flow is measured by instruments placed in the line to each ion-exchange train. The effluent from the "cation" sand vessel is routed back to the RCBT "A", as shown in Figure 3.3. The remaining SDS equipment and EPICOR II are not used for RCS water processing.

3.4.2 RCS Water Demineralization (cont'd)

Periodic sampling of the process stream will occur during the processing of a batch of water. At the completion of processing a batch, the contents of RCBT "A" will be sampled to determine acceptability for injections of this water into the RCS. If the water is within specification, it is injected into the RCS.

The types of samples to be taken at RCBT after letdown and prior to reinjection are shown in Table 3.2.

3.4.3 Leakage Detection and Processing

Each submerged vessel is located inside a secondary containment box that contains spent fuel pool water. During operation the secondary containment lid is closed. This lid is slotted to permit a calculated quantity of pool water to flow past the vessels and connectors. Pool water from the containment boxes is continuously monitored to detect leakage and is circulated by a pump through one of the two leakage containment ion exchangers. Any leakage which occurs during routine connection and disconnection of the quick-disconnects will be captured by the containment boxes, diluted by pool water, and treated by ion exchange before being returned to the pool.

3.4.4 Off Gas and Liquid Separation System

An off gas and liquid separation system collects gaseous and liquid wastes resulting from the operation of the water treatment system.

3.4.5 Sampling and Process Radiation Monitoring System

The sampling glove boxes are shielded enclosures which allow water samples to be taken for analysis of radionuclides and other contaminants. The piping entering the glove boxes permits the withdrawal of a volume limited amount of sample into a collection bottle. Cylinders are purged by positioning valves to permit the water to flow through them and return to a waste drain header and into the off gas separator tank. A water line connects to the sample line to allow the line to be flushed after a sample has been taken.

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

3.4.5.1 Sampling System

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

3.4.5.2 Process Radiation Monitoring System

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

3.4.5.3 Transuranic Element Monitoring

Filter and process train samples are being analyzed for isotopes of Uranium and Plutonium.

3.4.6 Ion Exchanger and Filter Vessel Transfer in the Fuel Storage Pool

Prior to system operation, ion exchanger and filter vessels are placed inside the containment boxes and connected with quick-disconnect couplings. When it is determined that a vessel is loaded with radioactive contaminants to predetermined limits as specified in the Process Control Program, the system will be flushed with low activity processed water. This procedure flushes away waterborne radioactivity, thus minimizing the potential for loss of contaminants into the pool water while decoupling vessels. Vessel decoupling is accomplished remotely. Vessels are transferred using the existing fuel handling crane utilizing a yoke attached to a long shaft. The purpose of this yoke-arm assembly is to prevent inadvertent lifting of the ion exchange bed or filter vessel to a height greater than eight feet below the surface of the water in the pool. This device is a safety tool that will mechanically prevent lifting a loaded vessel out of the water shielding and preclude the possibility of accidental exposure of operating personnel.

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

3.5 Zeolite Mixtures

The SDS ion exchangers will contain a uniform mixture of IONSIV-96 and LINDE-A ion exchanger media. These two zeolites were selected for their proven capabilities while processing Reactor Building Sump water to remove radionuclides. IONSIV-96 primarily removes the isotopes of Cesium and LINDE-A removes the isotopes of Strontium.

The ratio of loading the two types of ion exchanger media will be determined by experimental data to determine the optimum loading.

Periodic sampling of the process stream will be used to verify the performance of the ion exchange media. If necessary revisions will be made to the loading ratios if conditions warrant to achieve the proper decontamination factors. Verification of the performance of the ion exchange media will be made in accordance with the Process Control Plan.

3.6 Waste Produced

Based on operating experience processing the Reactor Building sump water, the useful life of a zeolite resin bed is in excess of 100,000 gallons of waste water processed. At this point the DF of the zeolite bed for Strontium goes to 1.

Table 3.1

BWST WATER DATA*

NUCLIDE	CONCENTRATION
H-3	9.02E-02
Co-60	1.30E-05
Cs-134	1.10E-04
Cs-137	3.70E-04
Cl	0.14 ppm
pH	5.95
B	3538 ppm
SO_4	< 5 ppm

*References 3 and 4

Table 3.2

RCBT WATER SAMPLING

RCBT LETDOWN SAMPLE	RCBT INJECTION SAMPLE
Gamma Scan	Gamma Scan
Gross Beta - Gamma	Gross Beta - Gamma
Sr-90	Sr-90
pH	pH at 77°F
Conductivity	Conductivity
Boron	Boron
Na	Na
Cl	Cl
Sulfates	Sulfates
H-3	H-3
	Oxygen
	Fluorides

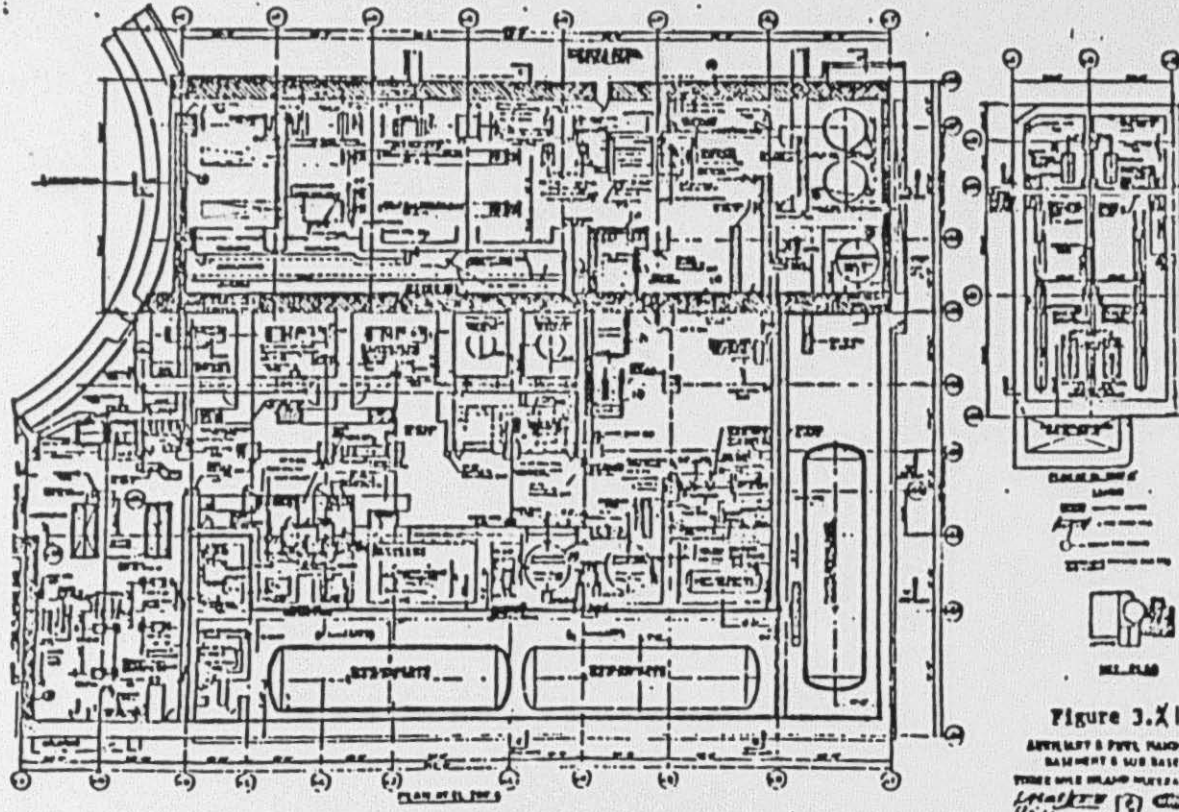


Figure 3.X1

AUXILIARY & FUEL HANDLING BUILDING
BASEMENT & SUB BASEMENT PLAN
THREE MILE ISLAND NUCLEAR STATION UNIT

3/82

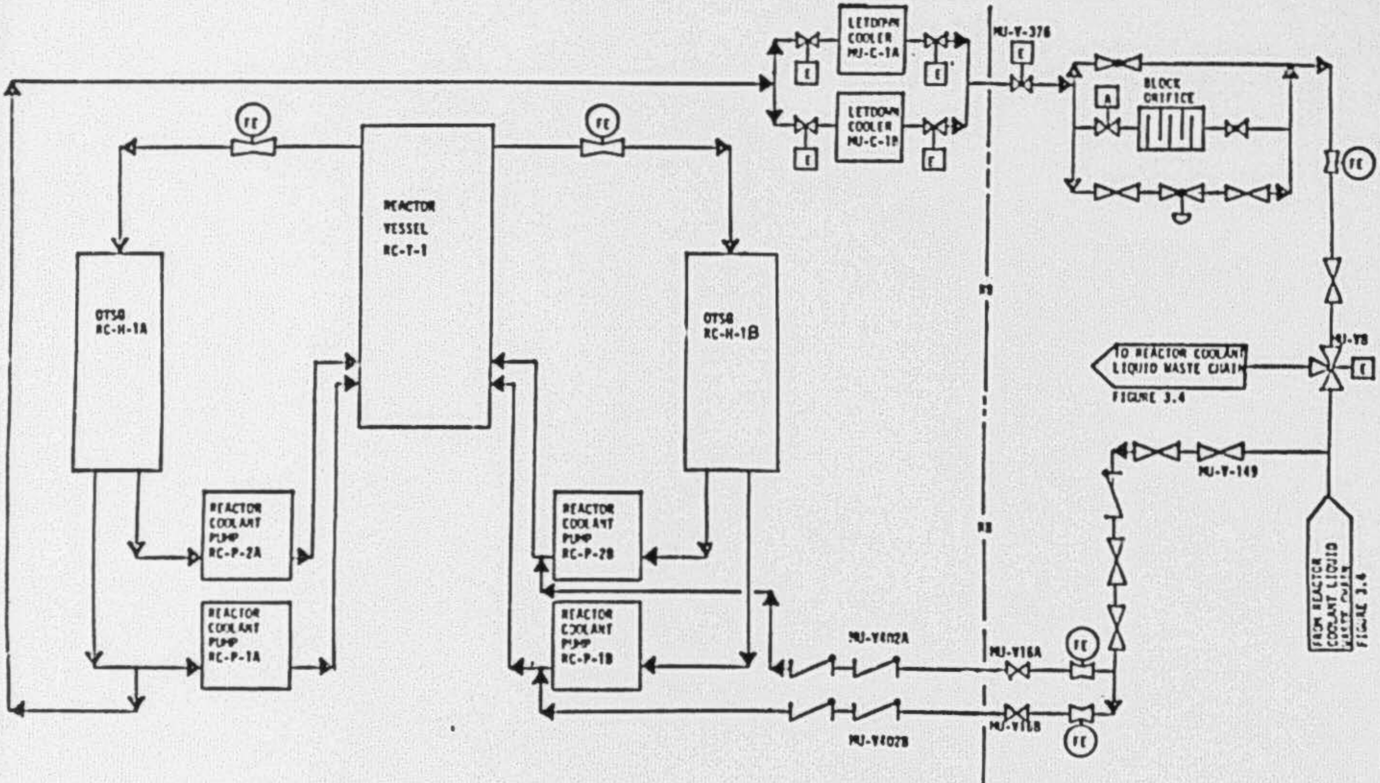


FIGURE 3.2
RCS LETDOWN AND INJECTION PATHS

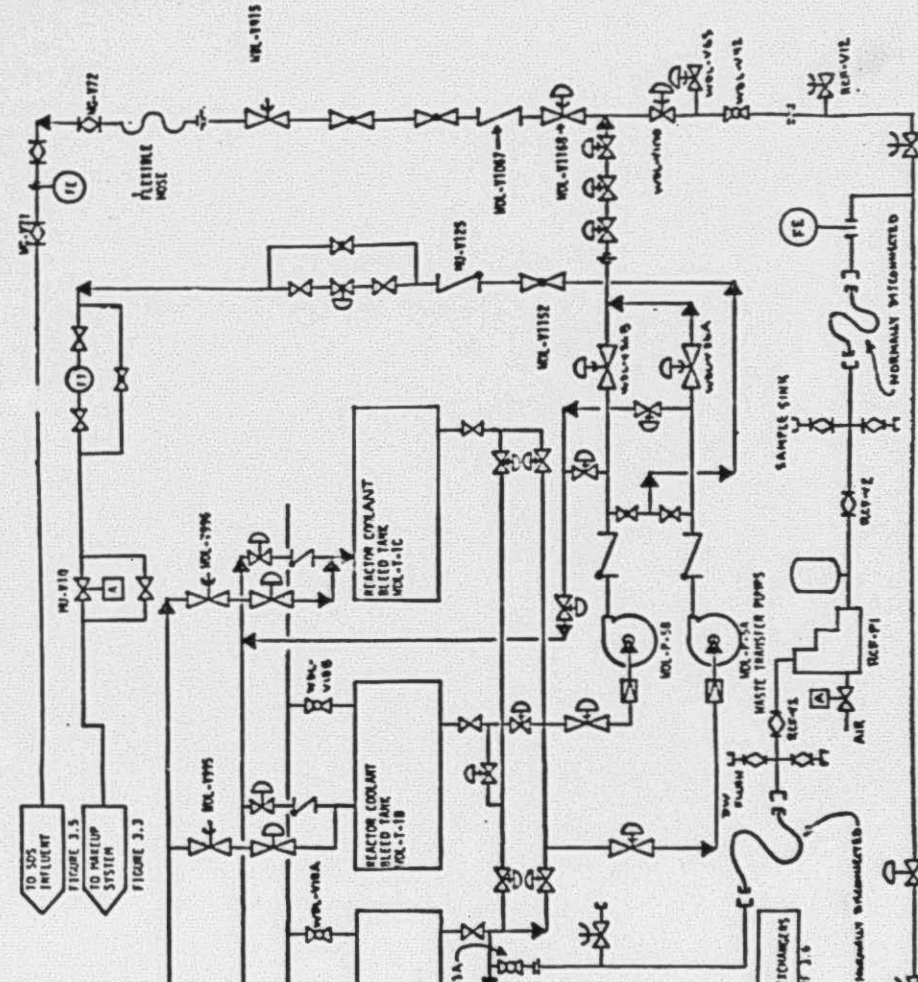


FIGURE 3.3
RCS PROCESSING COLLECTION STAGE
(REFERENCES 6, 7, 8, 9)

4.1 RCS Processing Safety Assessment

Processing of the RCS while in a partially drained condition does not present a unique safety concern. The actual processing of Reactor Coolant is adequately addressed in the SDS Technical Evaluation Report and the maintenance of the Reactor Coolant System in a partially drained condition is adequately addressed in the Quick Look Safety Evaluation. The only evolution not previously addressed is the simultaneous feed and bleed of the Reactor Coolant System in a partially drained configuration. During this evolution, RCS water level will be monitored and maintained by operating procedures. Such procedures will maintain the water level to within six (6) inches of the predetermined level set point. At the present RCS level, to permit incore inspections, this level is 210" $\pm$ 6". This level is the same as that established for the Quick Look program and will be monitored in a similar fashion. Thus this evolution will not increase the probability of occurrance or consequences of an accident previously evaluated or create the possibility of a different type accident, nor will the margin of safety as defined in the basis for any Technical Specification be reduced.

Appendix No. 2

to

Submerged Demineralizer System

Technical Evaluation Report

Title

Internals Indexing Fixure

Processing System

June 1983

CONTENTS

CONTENTS (continued)

Chapter 1

SUMMARY OF TREATMENT PLAN

1.1 Project Scope

Reactor Coolant System (RCS) processing via IIF capability must be continually available. The combined volume of water in the Rx Vessel & the internals indexing fixture (IFF) is 50,000 gal. The current RCS activity is given in Table 1.1. The primary purpose for the system, is the control of dissovled radio nuclides however, improvement of water clarity is expected as a byproduct.

This report describes the post Rx vessel head removal processing of the Reactor Coolant System (RCS) by the Submerged Demineralizer System (SDS) and other interfacing plant system for the maintainance of RCS dissoved radionuclide concentrations. The design features of this processing method are:

1. Use of the proven processing capabilities of the SDS.
2. Use of existing plant systems in support of the SDS.

This report is presented as an addendum to the previously submitted SDS Technical Evaluation Report (TER) (Reference 1) to provide greater detail in those aspects of system design and operation which are unique to the processing of the RCS & IIF water using the IIF Processing System.

1.2 Current RCS Radionuclide Inventory and Chemistry

Water samples have been taken weekly from the RCS to identify specific radionuclides and concentrations, and plant chemistry. Current results are listed in Table 1.1. This data is based on actual samples taken. RCS activity decreases due to radioactive decay, leakage from the RCS, and RCS processing. However, RCS activity may increase during processing shutdown due to leaching. Radionuclides concentrations are expected to be less in the IIF due to the dilution of the Rx Vessel when the IIF is filled with RCS grade processed water.

1.3 IIF Processing Description

Figure 1.1 shows a block diagram of the IIF processing flow path. On a batch basis, radioactive RCS water is pumped directly to the SDS and processed through the prefilter and final filter. RCS water then goes through the RCS manifold and both trains of ion exchangers at a flow rate of up to 15 gpm. The effluent from the ion exchangers is routed through a sand filter in the cation position to either the A or C RCBT for sampling. Concurrently, while water is being removed for processing, previously processed RCS grade water from the other bleed tank is being injected into the RCS to maintain water level. The return flow path is identical to that used during the prehead lift RCS processing.

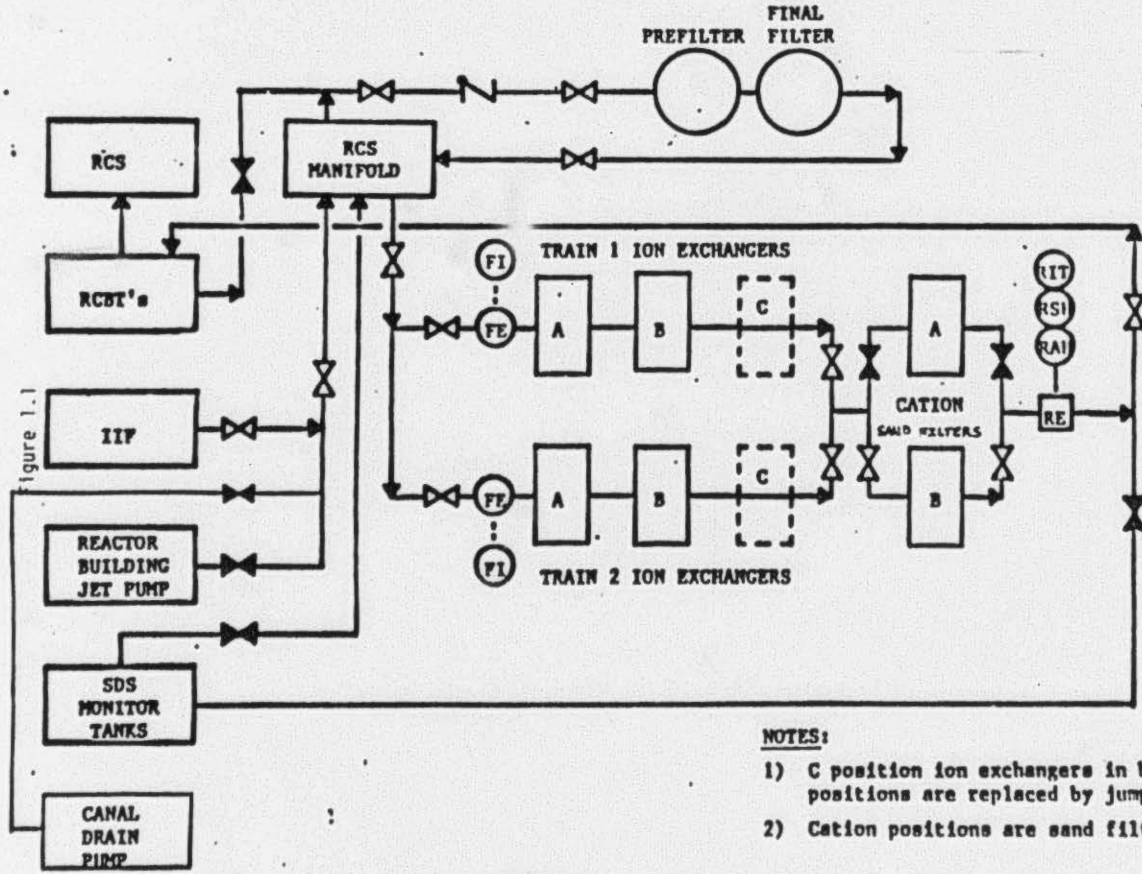
The IIF processing of the RCS will use the existing filter and ion exchangers of the SDS. Existing sampling connections will be used on the influent and effluent of the filters and ion exchangers to determine radionuclide and chemical composition of the process steam before and after processing.

As described in the SDS TER, the prefilters and final filters consist of sand filters for the removal of particulate matter. These filters are followed by a series of ion exchange vessels containing about 8 cubic feet of zeolite ion exchange media. Location, operation, and handling of these vessels remains unchanged from the mode of operation used for the prehead lift RCS processing. However both trains will be used. A sand filter in the cation position will replace the existing post filter in order to lower system differential pressure and improve effluent turbidity.

TABLE 1.1

RCS RADIONUCLIDE AND CHEMISTRY DATA
(06/15/84)

ISOTOPE	RADIONUCLIDE CONCENTRATION μCi/cc
H-3	0.045
Sr-90	2.7
Sb-125	0.14
Cs-134	0.015
Cs-137	0.3
Gross Beta	4.1
pH	7.72
Boron	5200 ppm
Na	1500 ppm
Cl	1.25 ppm
Turb	1.5 NTU



NOTES:

1) C position ion exchangers in both positions are replaced by jumpers.
2) Cation positions are sand filters.

FIGURE 1.1

IIF PROCESSING FLOW SHEET

Chapter 2

IIF PROCESSING PLAN DESIGN CRITERIA

2.1 Introduction

The IIF Processing Plan is designed to use a high capacity submersible pump (DWC-P-1) the Submerged Demineralizer System (SDS) and portions of existing plant liquid radwaste disposal systems to decontaminate the RCS water. This will reduce plant personnel and the possibility for off site radiation exposures. The design objectives of this processing plan are:

1. A system that is as independent as possible from existing radioactive waste systems at TMI-2. The SDS portion of this plan is a temporary system for the recovery of TMI-2. Only small sections of existing TMI-2 plant systems will be used.
2. A system that has proven performance in processing radioactive waste. The SDS portion of this processing plan has successfully decontaminated RCS water. Also, the reinjection path used for the IIF Processing Plan is the same path as was used for prehead lift RCS processing.

2.2 Design Basis

2.2.1 Submerged Demineralizer System

The Submerged Demineralizer System was designed in accordance with the following regulatory documents:

1. Code of Federal Regulations, 10CFR20, Standard for Protection against Radiation.
2. Code of Federal Regulations, 10CFR50, Licensing of Production and Utilization Facilities.
3. U.S. Regulatory Guide 1.21, dated June 1974.
4. U.S. Regulatory Guide 1.140, dated March 1978.
5. U.S. Regulatory Guide 1.143, dated July 1978.
6. U.S. Regulatory Guide 8.8, dated June 1978.
7. U.S. Regulatory Guide 8.10, dated May 1977.

The design basis for the SDS is presented in greater detail in Chapter 4 of the SDS Technical Evaluation Report.

2.2.2 Interfacing Systems

The interfacing systems with the SDS in the IIF Processing system are:

1. Reactor Coolant Liquid Waste Train.
2. Purification and Makeup System.

3. Auxiliary and Fuel Handling Buildings Heating Ventilation and Air Conditioning Systems.
4. Nitrogen Supply System.
5. Decay Heat Removal System.
6. Waste Gas System.
7. Standby Pressure Control System
8. Solid Waste Handling System
9. R.B. Jet Pump
10. Fuel Transfer Canal Drain System

The design criteria for these systems 1 through 6 above are presented in Chapter 3 of the TMI-2 FSAR. Conformance to these criteria is presented in the respective sectons for these systems in the TMI-2 FSAR. Item 7 is covered in the TMI Recovery System Descriptions and TER's. Items 8 through 10 will be included in the next revision of the SDS System Description.

2.3 IIF Processing Plan Goal

The goal of the IIF Processing Plan is to maintain the Cs137 radionuclide concentration in the RCS to less than 1 μCi/cc and reduce the Sr90 concentration. The RCS chemistry will be maintained as follows as a minimum:

Chlorides	<5 ppm
Oxygen	Atmospheric Concentration
pH	> 7.5 but < 8.4
Boron	> 4950 ppm

The processing of water through the SDS does not have any effect on the chemical characteristics of the RCS water. The chemistry specified above will ensure that there will be no adverse effects on the RCS with respect to corrosion. The boron concentration will also ensure that sufficient boron in present to maintain the core in a noncritical safe condition.

Chapter 3

SYSTEM DESCRIPTION AND OPERATIONS

3.1 Introduction

The IIF Processing Plan is designed specifically for the controlled decontamination of the radioactive water in the Reactor Vessel & IIF and the treatment of the radioactive gases and solid radioactive waste which are produced. This plan will use the SDS as the means of decontamination of the IIF with support from other existing plant systems and a new suction pump. See Fig. 1.1 for a diagram of the IIF processing system flowpath.

3.1.1 Submerged Demineralizer System

The SDS consists of a liquid waste processing system, an off gas system, a monitoring and sampling system, and solid waste handling system. The liquid waste processing system decontaminates the RCS water by a process of filtration and demineralization. The off gas system collects, filters and absorbs radioactive gases produced during processing, sampling, dewatering and spent SDS liner venting. The sampling system provides measurements of process performance. The solid waste handling system is provided for moving, dewatering, vacuum drying, inerrization, storage, and loading of filters and demineralizer vessels into the shipping cask.

3.1.2 Interfacing Systems

Rx coolant water is transferred from the IIF using a commercially available high capacity submersible pump. This pump (Indexing Fixture Processing Pump DWC-P-1) is supported from the IIF and takes suction approximately 2 1/2 ft above the Rx Vessel Flange. A 1 1/2 inch ID rubber hose with quick-disconnect two way shutoff type fitting connects the discharge of the pump to the fuel transfer canal drain manifold.

The manifold serves as a tie-in point for 3 systems; the Reactor Bldg. Basement Pump system, the fuel transfer canal drain system, and the IIF processing system. Double isolation of the IIF processing system from the other two is provided by air operated ball valve FCC-V003 and check valve FCC-V016 in addition to manual valves and disconnected/capped connections located on each of the other branches of the manifold. From the manifold, the system uses an existing flow path through Reactor Building penetration R-626, Fuel Handling Building penetration 1551 to tie-in and interface with the SDS system. Power for the pump is supplied from circuit 11 of distribtuion panel PDP-6A. Initial filling of the IIF is accomplished by transferring reactor coolant grade water to the IIF via the Fuel Transfer Canal Fill System, or via a waste transfer pump to an RCS cold leg.

Subsequent makeup to the IIF is accomplished by transferring reactor coolant grade water from RCBT-1A to the IIF via a waste transfer pump and an existing flow path through the WDL and MU systems to a cold leg of the reactor coolant system.

The roles of the RCBT's (1A & 1C) can be interchanged provided valves are properly realigned and the tank used to fill the IIF containes reactor coolant grade water.

Flow from the IIF may be manually throttled at valves CN-V-FL-1 or CN-V-FL-3 in SDS if desired. Flow to the IIF may be automatically controlled by valve MU-V9 based on IIF water level or manually controlled using WDL-V-167 and WDL-V-36A. Shutoff of the IIF supply (via WDL-V40) and discharge flows (via FCC-V003) is achieved automatically in the event of unacceptable water levels in the IIF and may also be manually accomplished at several locations.

The Fuel Handling Building, Auxiliary Building, and Reactor Building HVAC systems provide tempered ventilating air and controlled air movement to prevent spread of airborne contamination with the plant and to the outside environment. The Nitrogen Supply system provides N_2 for blanketing the Reactor Coolant Bleed Tanks. Reactor Coolant grade water currently contained in the RCBT's provides borated water for injection into the RCS/IIF for the initial fill operation. The Waste Gas System processes the gases from the vents from the RCBT's.

The Standby Pressure Control System, installed as a temporary TMI-2 recovery system, will be used as a safety system to ensure that a second RCS injection path is available.

The principal components of the SDS are located in Spent Fuel Pool "B", as shown in Figure 3.1. The piping and components of the systems interfacing with the SDS are located in the Fuel Handling and Auxiliary Buildings. Tanks, pumps, valves, piping, and instruments are located in controlled access areas. Components and piping containing significant radiation sources are located in shielded cubicles, such as the Reactor Coolant Bleed Tanks and the Waste Transfer pumps WDL-P-5A and WDL-P-5B (see Figure 3.2).

3.2 IIF Transfer Operations

3.2.1 Normal Operation

Under normal operating conditions the IIF will be filled to the desired water level and normal start up will be required. To start the IIF processing system the valves must be aligned per reference 8 (or the reverse alignment as discussed in paragraph 4, section 3.1.2), SDS must be configured for reactor coolant processing, the automatic trip switches must be in the not-blocked position, the surface suction system must be flushed, and both the surface suction system and the fuel transfer canal drain system must be isolated at the Fuel Transfer Canal Drain Manifold. The supply water to the IIF must be sampled to verify that it is within specification for reactor coolant grade water.

IIF processing system start-up is begun by starting waste transfer pump WDL-P-5B and opening valve WDL-V40. The pump will remain in minimum recirculation until flow to the IIF is required and MU-V-9 is opened. Valve FCC-V003 is remotely opened from SDS control panel CN-PNL-1, automatically starting the IIF processing pump to begin transferring water to the SDS. Valve MU-V9 automatically opens or is manually opened and maintains the IIF level.

3.2.2 Infrequent operation

The system has incorporated two hand switches (DWC-HIS-1A and DWC-HIS-1266-1) which are located on SDS control panel CN-PNL-1 which can be used to block automatic shutdown of the system for high or low levels. These switches will allow the operators to fill or drain the IIF to the desired water level needed.

Relocation of the IIF processing pump within the IIF may be required to avoid interferences with post head removal activities. By using flexible hose with two-way shutoff "quick disconnect" couplings on the pump discharge, and the overlapping hanger design for the pump support, movement of the pump can easily be accomplished using overhead material handling equipment.

3.3 IIF Instrumentation & Control

On/off controls for the waste transfer pumps are located on radwaste panel 301B, and in the control room on control panel 9.

Valve WDL-V40 has existing open/close controls located on radwaste panel 301B and in the control room on the control panel 9. Additional open/close controls are located on SDS control panel CN-PNL-1. WDL-V40 terminates flow in the event of high or low water level in the IIF. A block switch is located on CN-PNL-1 which can be used to block the low level trip to permit filling the IIF to the desired level.

Normal control of valve FCC-V003 and the IIF processing pump is performed from SDS control panel CN-PNL-1. A single hand switch controls operation of both the valve and pump. A dual indicating light is provided on CN-PNL-1 for valve position. FCC-V003 stops flow in the event of high or low water level in the IIF. A block switch is located on CN-PNL-1 which can be used to block the high level trip to permit draining the IIF to the desired level.

Water level will be maintained manually or automatically at a prescribed level (approximately 327'6") in the IIF by valve MU-V9. The control signal to valve MU-V9 is provided by the reactor water level monitoring system (bubbler) through proportional controller RC-LIC-102 or from RC-LT-102 which is located on control room panel SPC-PNL-3.

Emergency stop switches are provided at the IIF to close valves FCC-V003 and WDL-V40, thereby stopping flow to an from the IIF. (Closure of valve FCC-V003 will in turn trip pump DWC-P-1.) High and low level switches are provided on the bubbler panel to automatically stop flow to and from the IIF in the same manner as the emergency stop switches, and will sound alarms locally, in the control room on panel SPC-PNL-3 and at SDS panel CN-PNL-1.

IIF level indication is provided on panels SPC-PNL-3 and CN-PNL-1 and bubbler panel RC-LCP1.

3.4 IIF Processing by SDS

3.4.1 IIF Water Filtration

A flow diagram of the waste water filtering is shown in Figure 3.3. Two filters have been installed to filter out solids in the untreated contaminated water before the water is processed by the ion-exchangers. Both filters are sand type. The two sand filters are loaded in layers, using various sand sizings to optimize filter performance.

Mixed uniformly with the sand is approximately 6 pounds borosilicate glass which is at least 22 weight percent boron. The purpose of the borosilicate is to prevent the possibility of criticality should any fuel fines be transported in the letdown.

The flow capacity through each filter is 50 gpm. Reverse flow through filters is prevented by a check valve in the supply line to each filter.

Each filter is housed in a containment enclosure to enable leakage detection and confinement of potential leakage. The filters are submerged in the spent fuel pool for shielding considerations. Contaminated water is pumped (using DWC-P-1) through the filters and the RCS manifold to the ion exchangers.

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

Inlet, outlet, and vent connections on the filters are made with quick disconnect valved couplings which are remotely operated from the top of the pool. Inlet-outlet pressure gauges are provided to monitor and control solids loading. Load limits for the filters are based on filter differential pressure, filter influent and effluent sampling, and/or the surface dose limit for the filter vessel. A flush line is attached to the filter inlet to provide a source of water for flushing the filters prior to removal.

3.4.2 IIF Water Demineralization

A flow diagram of the ion exchange manifold and primary ion-exchange columns is shown in Figure 1.1. This system consists of eight underwater columns (24 1/2" x 54 1/2"), each capable of containing eight cubic feet inorganic zeolite sorbent. Homogeneously mixed Ion Siv IE-96 and LINDE-A zeolite are the medias of choice to efficiently immobilize the Cesium and Stronitum in the RCS. Four zeolite beds are divided into two trains each containing two beds (A, B) with piping and valves provided to operate either train individually or both trains in parallel. In order to maintain the IIF radiation levels ALARA, higher SDS Process flow rates are necessary. Normal operation will consist of two trains of two vessels per train operating in parallel with a flow rate of 5-15 gpm per train.

An in-line radiation monitor measures the activity level of the water exiting the last polishing filter. The valve manifold for controlling the operation of the primary ion exchange columns is located above the pool, inside a shielded enclosure that contains a built-in sump to collect leakage that might occur. Any such leakage is routed back to the RCS manifold via the off gas bottoms separater tank and pump. A line connects to the inlet of each ion exchanger to provide water for flushing the ion exchangers when they are loaded. Radionulcides loading of ion exchange vessels is determined by analyzing the influent and effluent from each exchanger. Process water flow is measured by instruments placed in the line to each ion-exchange train. The effluents from the two trains of ion exchangers is routed through either of the two "cation" positions. However for IIF processing, these "cation" positions will contain sand polishing filters similar to the pre and final filters instead of zeolite beds. Sand filters are used to improve effluent clarity. The existing post filter is bypassed to reduce system differential pressure. Therefore the zeolite effluent will pass through a sand filter in either of the cation positions directly to one of the RCBT's.

Periodic sampling of the process stream will occur during the processing of a batch of water. At the completion of processing a batch, the contents of RCBT "A" will be sampled to determine acceptability for injection of this water into the RCS. If the water is within specification, it is injected into the RCS.

The types of samples to be taken at RCBT after letdown and prior to reinjection is shown in Table 3.1.

3.4.3 Leakage Detection and Processing

Each submerged vessel in located inside a secondary containment box that contains spent fuel pool water. During operation the secondary containment lid is closed. This lid is slotted to permit a calculated quantity of pool water to flow past the vessels and connectors. Pool water from the containment boxes is continuously monitored to detect leakage and is circulated by a pump through one of the two leakage containment ion-exchangers. Any leakage which occurs during routine connection and disconnection of the quick-disconnects will be captured by the containment boxes, diluted by pool water, and treated by ion-exchange before being returned to the pool.

3.4.4 Off-Gas and Liquid Separation System

An off-gas and liquid separation system collects gaseous and liquid wastes resulting from the operation of the water treatment system. This system will be operated in the same manner for RCS water processing as it was for Reactor Building Sump water processing.

3.4.5 Sampling and Process Radiation Monitoring System

The sampling glove boxes are shielded enclosures which allow water samples to be taken for analysis of radionuclides and other contaminants. The piping entering the glove boxes permits the withdrawl of a volume limited amount of sample into a collection bottle. Cylinders are purged by positioning vavles to permit the water to flow through them and return to a waste drain header and into the off-gas seperator tank. A water line connects to the sample line to allow the line to be flushed after a sample has been taken.

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

3.4.5.1 Sampling System

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

3.4.5.2 Process Radiation Monitoring System

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

3.4.5.3 Transuranic Element Monitoring

During IIF processing the need for TRU Monitoring is not required, as the majority of TRU is expected to deposit in the filters & Ion Exchangers before reaching the RCBT.

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

Prior to system operation, ion exchanger and filter vessels are placed inside the containment boxes and connected with quick-disconnect couplings. When it is determined that a vessel is loaded with radioactive contaminants to predetermined limits as specified in the Process Control Program, the system will be flushed with low-activity processed water. This procedure flushes away waterborne radioactivity, thus minimizing the potential for loss of contaminants into the pool water while decoupling vessels. Vessel decoupling is accomplished remotely. Vessels are transferred using the exisiting fuel handling crane utilizing a yoke attached to a long shaft. The purpose of this yoke-arm assembly is to prevent inadvertent lifting of the ion exchange bed or filter vessel to a height greater than eight feet below the surface of the water in the pool. This device is a safety tool that will mechanically prevent lifting a loaded vessel out of the water shielding and preclude the possibilty of accidental exposure of operating personnel.

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

3.5 Zeolite Mixtures

The SDS ion exchangers will contain a uniform mixture of IONSIV-96 and LINDE-A ion exchanger media. These two zeolites were selected for their proven capabilities while processing Reactor Building Sump water to remove radionuclides. IONSIV-96 primarily removes the isotopes of Cesium and LINDE-A removes the isotopes of Strontium.

The ratio of loading the two types of ion exchanger media will be determined by experimental data to determine the optimum loading.

Periodic sampling of the process stream will be used to verify the performance of the ion exchange media. If necessary, revisions will be made to the loading ratios if conditions warrant to achieve the proper decontamination factors. Verification of the performance of the ion exchange medai will be made in accordance with the Process Control Plan for IIF processing.

TABLE 3.1

RCBT WATER SAMPLING

RCBT INJECTION SAMPLE

Gamma Scan
Gross Beta - Gamma
Sr-90
pH at 77°F
Conductivity
Boron
Na
Cl
Sulfates
H-3
Oxygen
Fluorides

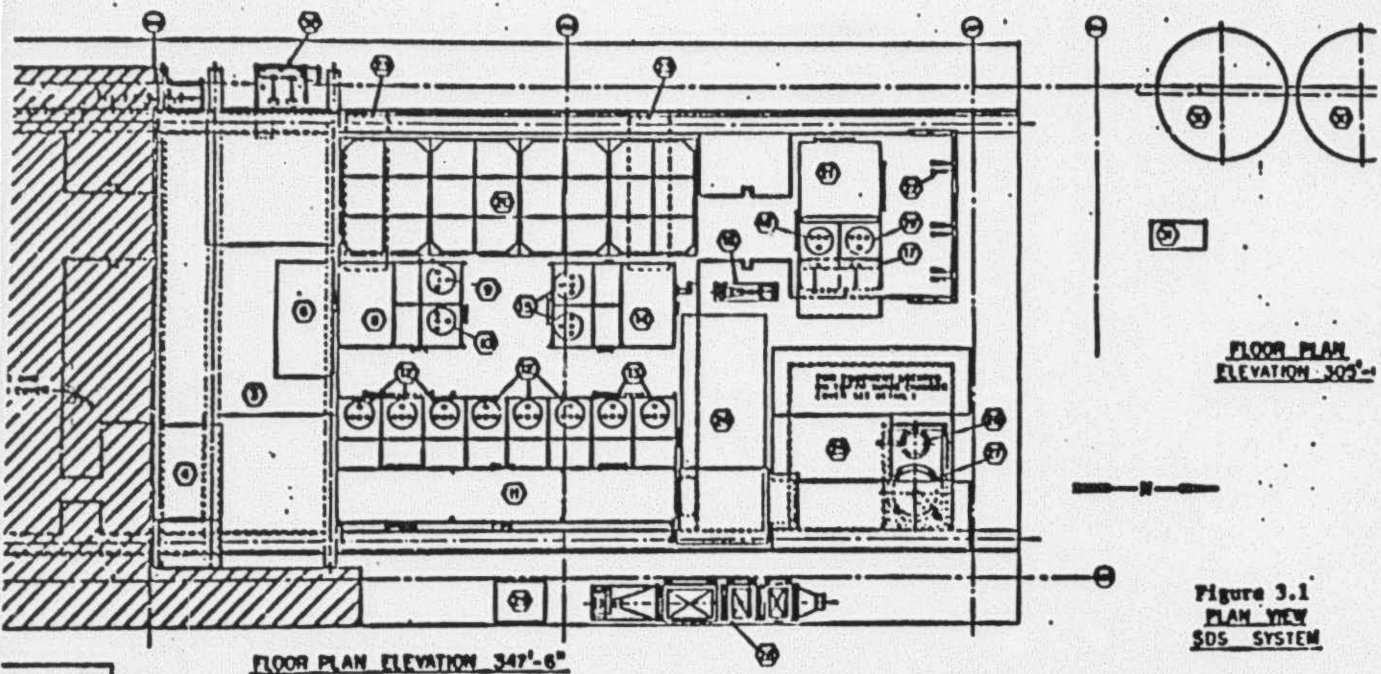
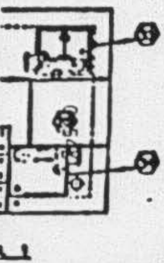


Figure 3.1
PLAN VIEW
SDS SYSTEM



1	
2	
3	CASK SUPPORT PLATFORM
4	RCS CLEAN-UP MANIFOLD
5	
6	FILTER MANIFOLD
7	
8	FILTER SUPPORT RACK
9	PRE-FILTER
10	FINAL FILTER
11	EXCHANGER SUPPORT RACK
12	PIN EXCHANGER
13	ION EXCHANGER

17	DEWATERING STATION SUPPORT RACK
18	ION EXCHANGER DEWATERING STATION
19	FILTER DEWATERING STATION
20	SPENT EXCHANGER STORAGE RACK
21	SHIPPING CASK SUPPORT PLATFORM
22	YOKE HANGER
23	MOVEABLE PERSONNEL BRIDGE
24	EXCHANGER MANIFOLD
25	SURGE CHAMBER COVER
26	OFF-GAS PUMP STAND PIPE
27	OFF-GAS SEPARATION
28	GAS FILTER UNIT
29	GAS SAMPLING
30	[illegible] TANK

33	HI-RAD FEED SAMPLE GLOVE BOX
34	INTERMEDIATE LEVEL SAMPLE GLOVE BOX
35	BETA MONITOR MANIFOLD
36	ANNUNCIATOR PANEL
37	RAD MONITOR PANEL
38	OFF-GAS SEPARATOR LEVEL INSTR. PANEL

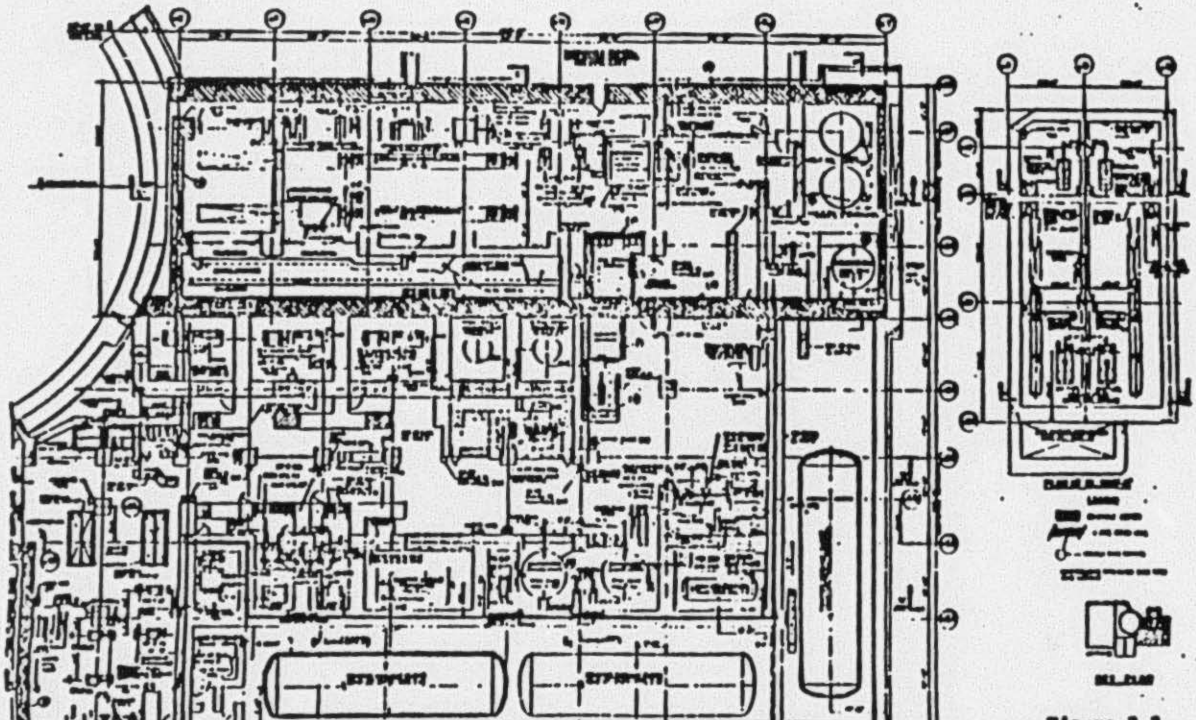
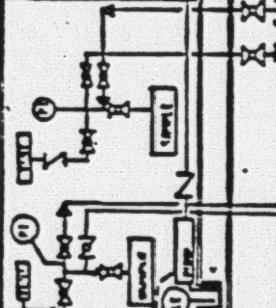
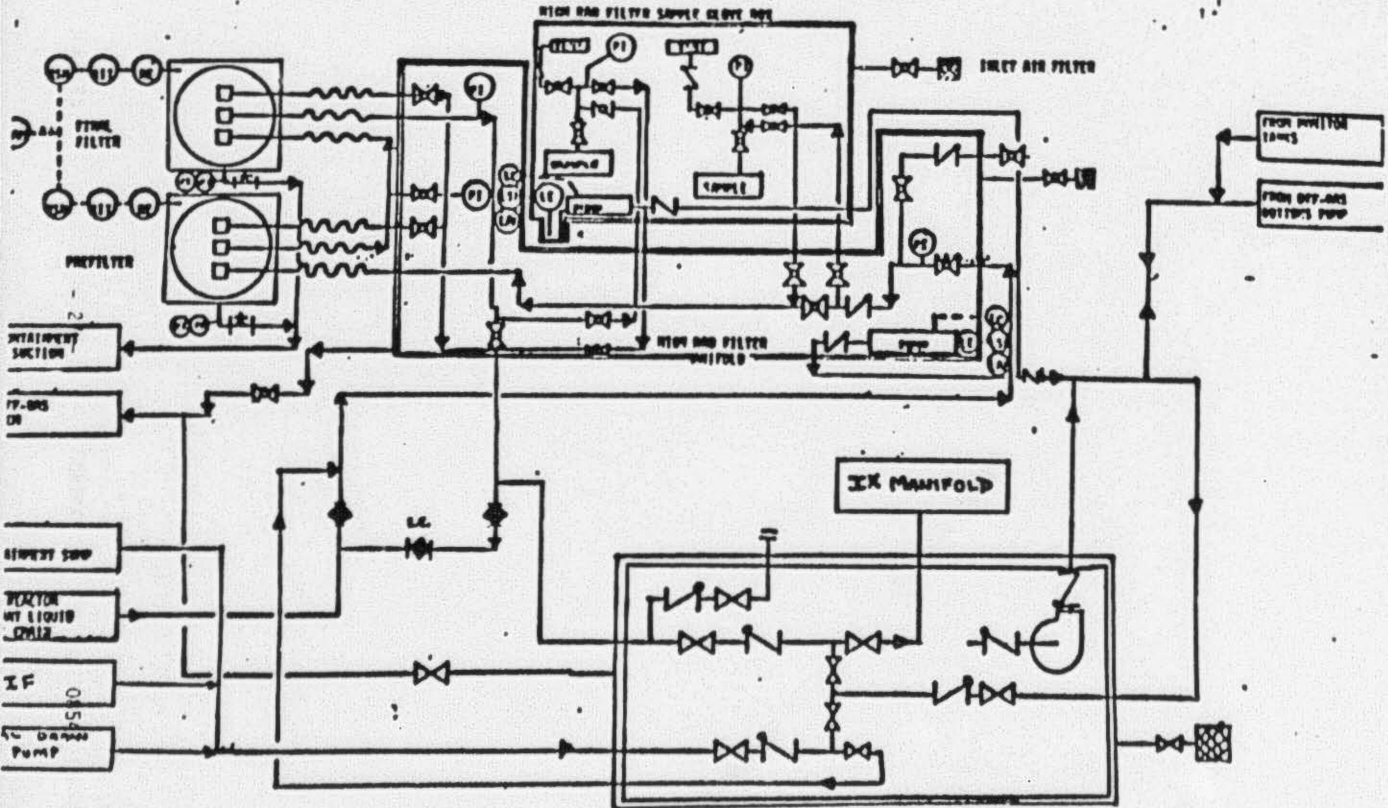


Figure 3.2

AUXILIARY & FUEL HANDLING BUILDING
BASEMENT & SUB BASEMENT PLAN
THREE MILE ISLAND NUCLEAR STATION UNIT 1





RCS Manifold

CHAPTER 4

Radiation Protection

4.1 Ensuring Occupational Radiation Exposures are ALARA

4.1.1 Overall Policy

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

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

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

4.1.2 SDS Design and Operation

The SDS design and operational considerations are given in Chapter 6 of the SDS TER. These design and operational considerations and features remain unchanged from this evaluation.

The radiation dose exposures to plant personnel during IIF processing will be lower due to the fact that the radionuclide concentration in the RCS water is significantly lower than those experienced during processing of Reactor Building sump water. The design basis for shielding the SDS equipment is to reduce radiation levels to less that 1 mrem/hr using the radionuclide concentration of 200 μCi/cc of predominately Cesium. The radionuclide concentration of Cesium in the RCS water is currently less than 0.5 μCi/cc.

4.1.3 Existing Plant Considerations

The radiation protection features for the existing plant system which interface with the SDS are described in Chapter 12 of the TMI-2 FSAR. The existing radiation shielding within the Auxiliary Building for the following systems is adequate to reduce the radiation levels to below the design basis of 2 mrem/hr in areas requiring access:

1. Makeup and Purification System
2. Reactor Coolant Liquid Waste Chain
3. Miscellaneous Waste Chain
4. Waste Gas System

4.2 Dose Assessment

4.2.1 On Site Assessment

Operation of the SDS in the IIF processing mode is expected to require 50,000 gallons of processing per week from installation of the IIF processing system until plenum removal (approximately 14 months). This amount of processing is required to maintain Cs-137 concentration at less than 1.0 μCi/ml in order to maintain radiation levels in the reactor vessel head area as low as reasonably achievable. Based on current experience with the SDS this amount of processing is expected to result in an exposure for SDS operating area activities of 0.225 man-rem/week or approximately 13.5 man-rem over the 14 month period.

4.2.2 Off-site Radiological Exposures

Source Terms for Liquid Effluents

Liquid effluent from the system will be returned to station tankage for further disposition. Therefore, no liquid source term is identified for this evaluation.

Source Terms for Gaseous Effluents

The plant vent system is a potential pathway for carrying airborne radiactive material and release. Radionuclides in the gaseous effluent arise from entrainment during transfer of contaminated water to various tanks, filters, ion exchange units, and also from water sampling. For further information, see section 6.3.2 of the SDS TER.

Chapter 5

CONDUCT OF OPERATIONS

5.1 System Performance

By processing the Reactor Building sump water and RCS successfully assurance has been granted that components developed specifically to meet the conditions imposed at TMI will perform in the intended manner.

The ion-exchange process is a well understood process. The SDS has demonstrated that high decontamination factors can be achieved by the use of zeolite ion exchange media.

During IIF processing, the SDS system flow rates will be higher than during all previous processing. An eight hour test was performed to assure that these increased flowrates will not adversely affect zeolite performance. Also, calculations have been performed by ORNL to demonstrate that system performance will not be jeopardized. Although radionuclide break through may occur sooner in the batch, it will progress more slowly. This breakthrough will be allowed to occur to extend zeolite life (minimize wastes) since the effluent is routed back to the IIF.

Zeolite media loading and dewatering can be accomplished in the intended manner and remote tools, necessary for the coupling and de-coupling of the vessels, operate in the intended manner.

5.2 System Testing

Prior to use in the SDS each vessel will be hydrostatically tested in conformance with the requirements of applicable portions of the ASME Boiler and Pressure Vessel Code. Upon completion of construction, the entire system was pneumatically tested to assure leak-free operations. The system will be retested prior to IIF processing at the design pressure.

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

5.3 System Operations

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

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

Chapter 6

ADDITIONAL ACCIDENT SCENARIOS

6.1 Possible Accident Scenarios

6.1.1 Overfill of the IIF could result in contamination of the pool area in the vicinity of the reactor vessel, and potentially increase airborne activity in the Reactor Building.

6.1.2 Lowering of the IIF water level would reduce the shielding provided thereby increasing worker exposure on the IIF platform.

6.1.3 Injection of below specification borated water into the IIF would violate technical specification requirements, and could lead to criticality.

6.1.4 A breech of the system pressure boundary during pumping of the reactor coolant to the RCBT could result in the release of reactor coolant which could cause additional contamination of reactor building or Aux/FHB surfaces.

6.1.5 If FCC-V003 remains open after pump DWC-P-1 stops, siphoning from the IIF would result in lowering the IIF water level.

6.2 Design Features to Mitigate Effects of Accident Events

6.2.1 When transferring water from the RCBT to the IIF, flow rate will be automatically controlled by MU-V9, or manually controlled using WDL-V-36A and WDL-V-167. Valve WDL-V40 will receive input from a hi-level switch, which will automatically close the valve in the event of high water level, terminate system operation, and sound local and remote alarms. Additionally, when initially filling the IIF, the water level will be monitored by visual inspection to prevent overfilling of the IIF.

6.2.2 When transferring reactor coolant from the IIF through SDS to the RCBT, reactor coolant will be pumped using DWC-P-1. Suction will be taken about 2 1/2 feet above the reactor vessel flange. This will allow a sufficient inventory of reactor coolant to remain in the reactor vessel for shielding as well as decay heat removal in the event the system does not automatically stop for low level. Valve FCC-V003 will automatically close and alarms sound in the event of low level. Closing valve FCC-V003 automatically trips the IIF processing pump and terminates system operation. An emergency stop switch provided at the IIF will allow workers to terminate system operation in the event of low level indication.

6.2.3 System procedures will require that the surface suction system be flushed and both the surface suction system and the fuel transfer canal drain systems isolated before starting the IIF processing system. Double isolation is provided to prevent injection of below specification borated water into the IIF from these sources. Sampling of the RCBT supplying the IIF is required before the system is placed in operation.

6.2.4 The discharge hose and pipe, including the manifold will be leaked tested in accordance with ANSI B31.1. Periodic visual inspection of the hose is required to assess its condition. The hose and couplings are rated higher than the IIF processing pump shutoff head (≈ 150 psig).

6.2.5 FCC-V003 is provided with a spring loaded actuator which will cause the valve to fail close on loss of air or electricity, thereby preventing siphoning from the IIF. Should the valve remain open, suction will be broken 2 1/2 feet above the reactor vessel flange, leaving a sufficient inventory of reactor coolant for shielding and decay heat removal. Should water level drop dramatically personnel will be restricted from the immediate area.

REFERENCES

1. MET ED letter LL2-81-0070 dated March 11, 1981, G. K. Hovey (MET ED) to L. Barrett (NRC), "Three Mile Island Nuclear Station, Unit 2, Operating License No. DPR-73, Docket No. 50-320, Submerged Demineralizer System."
2. TMI-2 Radiochemistry Summary Sheet, Sample No. 84-08125 dated June 15, 1984.
3. TMI-2 Burns and Roe Drawing No. 2024, Makeup and Purification System.
4. TMI-2 Burns and Roe Drawing No. 2027, Radwaste Disposal Reactor Coolant Liquids.
5. TMI-2 Burns and Roe Drawing No. 2045, Radwaste Disposal Miscellaneous Liquids.
6. GPUNC Drawing No. 2R-950-21-001, P&ID Composite: Submerged Demineralizer System.
7. TMI-2 Burns and Roe Drawing No. 2026, Decay Heat and Spent Fuel Cooling.
8. TMI-2 Bechtel Drawing No. 2-MTS-DWC01 & 2, Schematic Diagram: IIF Processing System.