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PDR

July 9, 1980
TLL 283

TMI Program Office
Attn: Mr. John T. Collins, Deputy Director
U. S. Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, Pennsylvania 17057

Dear Sir:

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

Your letter NRC/TMI-80-89, dated May 16, 1980, requested additional information to enable your evaluation of the subject report.

Enclosed please find our responses to your comments. We are prepared to discuss these responses with you or your designated representatives at your convenience.

Sincerely,

G. K. Hovey
Director, TMI-II

GKH:LJL:dad

Enclosures

cc: B. J. Snyder

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GENERAL COMMENTS

QUESTION 1

Provide (1) a complete set of piping and instrument drawings (P&ID), (2) general arrangement drawings, (3) zeolite, cation, and filter vessel internal drawings, and (4) the system description document to the latest revision for SDS.

RESPONSE

Piping and instrument drawings (P&ID) and general arrangement (GA) drawings are enclosed in accordance with the attached listing of SDS drawings. This listing includes the current revision number. Zeolite vessel internal drawings are enclosed. Both the zeolite and cation vessel internals are of the same design. The SDS system description document is currently being internally reviewed. Upon completion of that review, on or about July 15, 1980, the SDS system description will be forwarded.

SUBMERGED DEMINERALIZER SYSTEM DRAWINGS

527D-A-5001, Rev. 03	Supply Manifold, Submerged Ion Exchanger P & ID
527D-A-5002, Rev. 02	Contaminated Feedwater System P & ID
527D-A-5003, Rev. 02	Monitoring Tank System P & ID
527D-A-5004, Rev. 03	Offgas and Liquid Separation System P & ID
527D-A-5005, Rev. 01	Ion Exchange Flow Sheet
527D-A-5006, Rev. 02	Intermediate Sampling System
527D-A-5007, Rev. 00	General Layout Plan - Ion Exchange Equipment
527D-A-5008, Rev. 00	Utility Flow Diagram
527D-A-5009, Rev. 01	Beta Monitoring Manifold P & ID
527D-A-5010, Rev. 00	Ion Exchanger Supporty Assembly
527D-A-5011, Rev. 01	Dewatering Station P & ID
527D-A-5012, Rev. 00	Pump Removal Instruction for Offgas Pump Feed System
527D-A-5013	RCS Clean-up Manifold P & ID (SEE NOTE 1)
527D-A-5014, Rev. 00	Pool Floor Equipment Base Location Installation Dimensions
527D-D-5002, Rev. 01	Prefilter, 125 Micron
527D-D-5003, Rev. 02	Final Filter, 10 Micron
527D-D-5004, Rev. 00	Prefilter/Final Filter Details
527D-D-5006, Rev. 00	Ion Exchanger with Agitator (SEE NOTE 2)

NOTE 1: This drawing is currently in development and will be furnished when available.

NOTE 2: This drawing subject to change pending resolution of solidification requirements.

GENERAL COMMENTS

QUESTION 2(a)

Provide a complete process flow diagram indicating (a) expected radioactivity concentration (µCi/ml) in each process stream including the final processed water for each radionuclide present...

RESPONSE

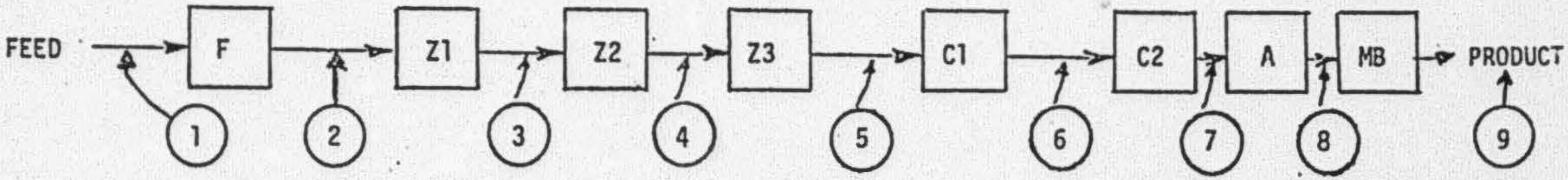
Oak Ridge National Laboratory has been tasked with the testing of the SDS process flow stream performance characteristics. These performance characteristics include the determination of process stream node point radionuclide concentrations. Attached to this response are the results of testing for the determination of radionuclide concentrations at process stream node points as submitted to GPU by ORNL on April 3, 1980. These results characterize the SDS process stream as follows:

F - pre-filter and final filter
Z1 - first zeolite bed
Z2 - second seolite bed
Z3 - third zeolite bed
C1 - cation bed
C2, A, MB - polishing bed

The model used for analysis of the process flow stream treats the polishing bed as though it is a combination of a separate cation bed, anion bed, and mixed resin bed. While it is physically possible to provide final polishing of the effluent flow stream in this manner, there are many other methods that can be employed to accomplish the objective of the polishing bed. Present plans for the SDS polishing bed specify that the resins will be loaded into one vessel and that they will be "layered" to achieve the final polishing function.

Radionuclide concentrations at the defined node points, as determined during the testing program, are given. These results are based on preliminary testing and the final report from ORNL will be submitted to the NRC in August, 1980, depending on receipt by GPU by July 30, 1980.

TABLE 4. EXPECTED ACTIVITY CONCENTRATIONS[a] IN SDS PROCESS STREAMS AFTER 200 BED VOLUMES THROUGH EACH ZEOLITE BED



	1	2	3	4	5	6	7	8	9	PROD. CONC. X 10CFR20[f]
^{3}H	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	333
^{60}Co	c	c	5 E-5	5 E-5	5 E-5	5 E-5	d	5 E-5	5 E-5	0.1
^{89}Sr	5.3E-1[b]	5.2E-1	2.0E-3	2.2E-4	6.9E-5	5.9E-5	d	1.5E-5	1.1E-5	3.7
^{90}Sr	2.32[b]	2.26	8.6E-3	9.4E-4	3.0E-4	2.6E-4	d	6.7E-5	4.8E-5	160
^{95}Nb	c	c	1.9E-5	1.0E-5	1.0E-5	1.0E-5	d	1.0E-5	1.0E-5	0.1
^{103}Ru	c	c	2.3E-5	1.5E-5	1.1E-5	1.0E-5	d	2.3E-6	1.5E-6	0.02
^{106}Ru	c	c	2.1E-3	1.4E-3	1.0E-3	9.2E-4	d	2.1E-4	1.4E-4	14
^{125}Sb	c	c	1.9E-2	1.9E-2	1.9E-2	1.9E-2	d	1.9E-2[e]	1.9E-2[e]	192
^{134}Cs	2.62E+1	2.62E+1	2.9E-3	7.0E-6	6.6E-6	6.6E-6	d	6.6E-6	3.0E-6	0.03
^{137}Cs	1.56E+2	1.56E+2	1.7E-2	4.2E-5	3.9E-5	3.9E-5	d	3.9E-5	1.8E-5	0.9
^{144}Ce	c	c	5.1E-4	3.9E-4	2.6E-4	1.9E-4	d	1.3E-4	6.4E-5	6.4

[a] In μCi/ml as of July 1, 1980.

[b] Difference in strontium concentration based on estimate of 1000 gal of solids in 650,000 gal of water.

[c] Not detected.

[d] Not yet determined.

[e] Concentration lower initially (until C1 and C2 beds became loaded with sodium).

[f] Multiples of limits of 10CFR20, Appendix B, Table II, Column 2.

GENERAL COMMENTS

QUESTION 2(b)

Provide a complete process flow diagram indicating (b) expected concentrations (ppm or ppb) of nonradioactive chemicals in each process stream. Include test results, references, and any supporting documents to support and substantiate the decontamination factors you assumed.

RESPONSE

Based on ORNL spark source mass analysis, the major nonradioactive chemicals in the SDS feed are estimated to be:

B	-	2200 ppm
Cl	-	15 ppm
Ca	-	10 ppm
Na	-	1200 ppm

These concentrations will be present in the process stream at the influent to the cation bed. Some Ca and Na will be removed by the cation mixed bed. Concentrations of these nonradioactive chemicals in the cation effluent/polisher influent will be:

B	-	2200 ppm
Cl	-	15 ppm
Ca	-	0-10 ppm
Na	-	0-1200 ppm

The polishing unit will consist of anion, cation, and mixed resin. Bed loadings, yet to be determined, will not be designed to deionize the influent water. It should be noted that the SDS design has been performed with the primary objective of removal of radioactive contaminants from the process flow stream. Any non-radioactive element removal from the flow stream is incidental to the process.

GENERAL COMMENTS

QUESTION 3

Provide expected radioactivity loading (Ci) and throughput (gallons) in each pre-filter, final filter, zeolite bed, cation bed, and polishing demineralizer.

RESPONSE

To bound the filter loading estimate, two approaches were considered. The first estimate (worst case) is based on ORNL report dated September 12, 1979. Thise case assumes that: (1) filter changeout occurs after 50,000 gallons throughput, (2) all solids are loaded on one vessel, (3) no decay of Sr 89 since available data does not distinguish between Sr 89 and Sr 90 for solids. The results of this approach indicates a worst case radioactivity loading of 554 curies on a vessel, of which 525 curies is Sr 89/90. The second estimate (realistic case) assumes that: (1) Strontium ratios in the solids are the same as in solution, (2) the feed (solids) would contain approximately 1.3×10^{-2} μCi/ml Sr 89 and .18 μCi/ml Sr 90 as of October 1, 1980. These assumptions yield filter loadings of 2.5 curies Sr 89, 34 curies Sr 90, 1.4 curies Cs 134, 14 curies Cs 137, and 4.7 curies Sb 125.

The zeolite beds are planned to be operated for approximately 200 bed volumes prior to changeout of the first bed and movement of beds 2 and 3 to positions 1 and 2, respectively. This intended plan of operation yields a bed throughput of approximately 12,000 gallons, based on a bed volume of 8 ft^3. The Technical Evaluation Report identifies that the zeolite bed volume is 7 ft^3. Recent testing has indicated that optimum zeolite bed loading is achieved when the bed is filled with 8 ft^3 of zeolite resins.

Studies at ORNL have indicated essentially all the cesium will be deposited on the first and second zeolite beds. Utlizing concentrations of 28 and 160 μCi/ml for Cs 134 and Cs 137 respectively, the first bed would contain an estimated 1270 curies of Cs 134 and 7270 curies of Cs 137. Beds two and three would contain a slightly higher level of activity due to being in service for 400 and 600 bed volumes respectively. ORNL column tests indicated that if feed solution contains 5.2×10^{-10} μCi/ml and 2.26 μCi/ml of Sr 89 and Sr 90, respectively, the first bed will contain approximately 23 curies of Sr 89 and 102 curies of Sr 90. Smaller amounts of other isotopes such as Nb 95, Ru 106, and Ce 144, will also be present.

RESPONSE TO QUESTION 3 (cont'd.)

Based on ORNL column testing, the cation bed in service will contain approximately 3 mCi of Sr 89/90 per 15,000 gallons of processed water. The polishing unit is estimated to contain the following per 15,000 gallons of process water:

ISOTOPE	mCi
Sr 89	2.7
Sr 90	12.0
Ru 103	.48
Ru 106	44.3
Cs 134	.2
Cs 137	1.2

It should be noted that zeolite column loading is based on 8 ft^3 of zeolite resins in the column.

GENERAL COMMENTS

QUESTION 4

Provide expected solid loading capacity for pre and final filters and estimate number of spent filter vessels to be generated as a result of SDS operation.

RESPONSE

The design of the pre-filter requires the use of a 125 micron filter element with 40 ft^2 of filter area. The final filter design requires the use of a 10 micron filter element with 80 ft^2 of filter area. Load testing to determine SDS filter loading has been performed by AGNS using a synthetic slurry similar to the containment sump water. This testing confirmed that the majority of the solids will be deposited on the final filter with only minor pre-filter loading. Assuming that the final filter loading is 1/2 pound per square foot and a total of 1000* pounds of filterable solids pass through these filters, it is expected that approximately twenty-five final filters will be expended. It is also probable that one pre-filter will be replaced after the initial startup of the SDS in order to maintain a lower DP across the two filters.

* NOTE: The 1000 pounds of filterable solids is based on 10 percent of the estimated solids on the reactor building containment floor being processed by the SDS.

GENERAL COMMENTS

QUESTION 5

Provide expected number of spent resin vessels (zeolite, cation, and pool purification mixed-bed) to be generated during operation of the SDS.

RESPONSE

Assuming that 200 bed volumes, approximately 15,000 gallons will be processed prior to zeolite bed removal and rotation, 67 zeolite beds will be expended to process 1,000,000 gallons of containment sump water. Based on column testing, ORNL has estimated that SDS operation would generate approximately 1,000 gallons of cation resin. This would expend approximately 17 cation beds. Assuming normal leakage, 2 to 4 pool purification vessels are anticipated to be utilized.

GENERAL COMMENTS

QUESTION 6

Describe process flow and method for processing the reactor coolant through the SDS and/or EPICOR-II.

RESPONSE

The SDS is currently designed to interface with the Reactor Coolant System through the MDHR system. The processing method to be used for the decontamination of RCS water is similar to that method planned for the containment sump water.

The processing of reactor coolant system water by EPICOR-II is specifically prohibited by the Memorandum and Order issued by the Nuclear Regulatory Commission on October 16, 1979. However, NUREG 0591, Environmental Assessment on the Use of EPICOR-II at TMI-II, states, "Waste water that is acceptable for processing in the EPICOR-II system is that which has Iodine-131 and Cesium-137 concentrations of less than 100 µCi/ml (intermediate level waste)". Because the RCS water has Cesium-137 concentrations that are below the maximum permissible EPICOR-II influent concentrations, we are evaluating the use of the EPICOR-II system for processing RCS waste water.

GENERAL COMMENTS

QUESTION 7

Describe the design and operational features of SDS polishing demineralizer.

RESPONSE

The design and operational features of the SDS polishing demineralizer has been identified as proprietary information under the provisions of 10 CFR 2.790. Chem-Nuclear Systems, Inc. will forward design details under separate cover.

GENERAL COMMENTS

QUESTION 8

Provide the maximum expected tritium concentration in the TMI-II fuel pool water with anticipated routine leakage (coupling and decoupling operations of filter and resin vessels) as a result of SDS operation.

RESPONSE

Recent testing on the 1½" "Hanson" connectors had indicated that a maximum of 100 ml of liquid will escape during the uncoupling operation. The estimated contribution of tritium in the pool is calculated to be 1×10^{-4} μCi/ml utilizing the following assumptions:

1. Pool volume - 233,450 gallons
2. Leakage tritium concentration - 1 μCi/ml
3. Three zeolite beds and one cation are uncoupled each 15,000 gallons
4. Two filters are uncoupled each 50,000 gallons
5. Three disconnects per vessel are required
6. Containment waste water is 1,000,000 gallons

The above calculations assume that processed water from the monitoring tanks is utilized for the flush sequence prior to disconnect.

Furthermore, it is anticipated that the spent fuel pool will be initially filled with processed water that contains a tritium concentration of approximately 1 μCi/ml. The contribution to existing pool tritium concentrations from uncoupling operations is insignificant.

GENERAL COMMENTS

QUESTION 9

Provide expected fuel pool water temperature, water volume, evaporation rate, and area ventilation flow rate, and resulting airborne tritium concentration in the TMI-I and TMI-II Fuel Handling Areas during operation of the SDS.

RESPONSE

If all the Cs 134 and Cs 137 contained in 1,000,000 gallons of containment water were deposited in vessels and stored in the "B" pool, and all decay energy is released to the pool, the rate of energy release is approximately 4000 watts. Therefore, the pool temperature is estimated to be approximately ambient building air temperature, 80°F.

The pool water volume is approximately 233,000 gallons and will have an estimated evaporation rate between 25 to 50 gallons per day.

The Fuel Handling Building exhaust rate is 36,000 cfm. 16,200 cfm are exhausted from the fuel storage pool area.

Calculations have shown that if processed water containing 1 μCi/ml tritium is used to fill the pool, the resulting airborned tritium concentrations could range between 3×10^{-10} μCi/ml to 6×10^{-10} μCi/ml. As indicated in Question 8, the additional tritium losses to the pool due to uncoupling operations would contribute an insignificant amount to increased spent fuel tritium, and therefore, will not significantly increase airborne tritium concentrations.

It is to be noted that the indicated airborne concentrations will exist only on the TMI-II side of the Fuel Handling Area. The TMI-I and TMI-II Fuel Handling areas will not communicate as a result of modifications being performed to accomplish unit separation prior to restart of TMI-I.

GENERAL COMMENTS

QUESTION 10

Describe the provisions provided for the removal of spent resin from the SDS polishing unit demineralizer.

RESPONSE

The SDS polishing unit demineralizer is a disposable unit. Our present intention is to dispose of the polishing unit at a shallow-land burial site in accordance with applicable regulations.

SPECIFIC COMMENTS

QUESTION 1

State if specific process features include processing of rinse water through the SDS. If so, estimate the volume and radioactivity concentration of rinse water.

RESPONSE

Flush lines on each of the submerged SDS components are connected to a drain header as indicated on P & ID 527D-A-5001. This drain line is connected to a shielded mist eliminator tank that drains to the offgas pump standpipe. An offgas bottoms pump is utilized to transfer liquid back to the SDS feed standpipe for reprocessing.

Concentrations within the mist eliminator tank will vary from a small fraction of, to nearly equivalent to, the SDS feed stream depending on operating conditions. It is estimated that approximately 200 gallons of flush water per submerged vessel will be utilized during a flush sequence.

SPECIFIC COMMENTS

QUESTION 2

Based on past experience, project the Reactor Building water level for the next two years and list the critical instruments that may be affected by the water level you projected.

RESPONSE

2.1 The projected water level in the Reactor Building for July 1, 1982 is 291.7 feet or a depth of 9.2 feed from the RB floor. This projection is based on the current water depth of 7.7 feet (as of May 15, 1980) with an inflow leakage of ∿ 0.1 gpm from the Reactor Coolant System (RCS). This inflow leakage rate has been detected during the period of time that the RCS has been maintained at a pressure of ∿ 100 psi (DVM).

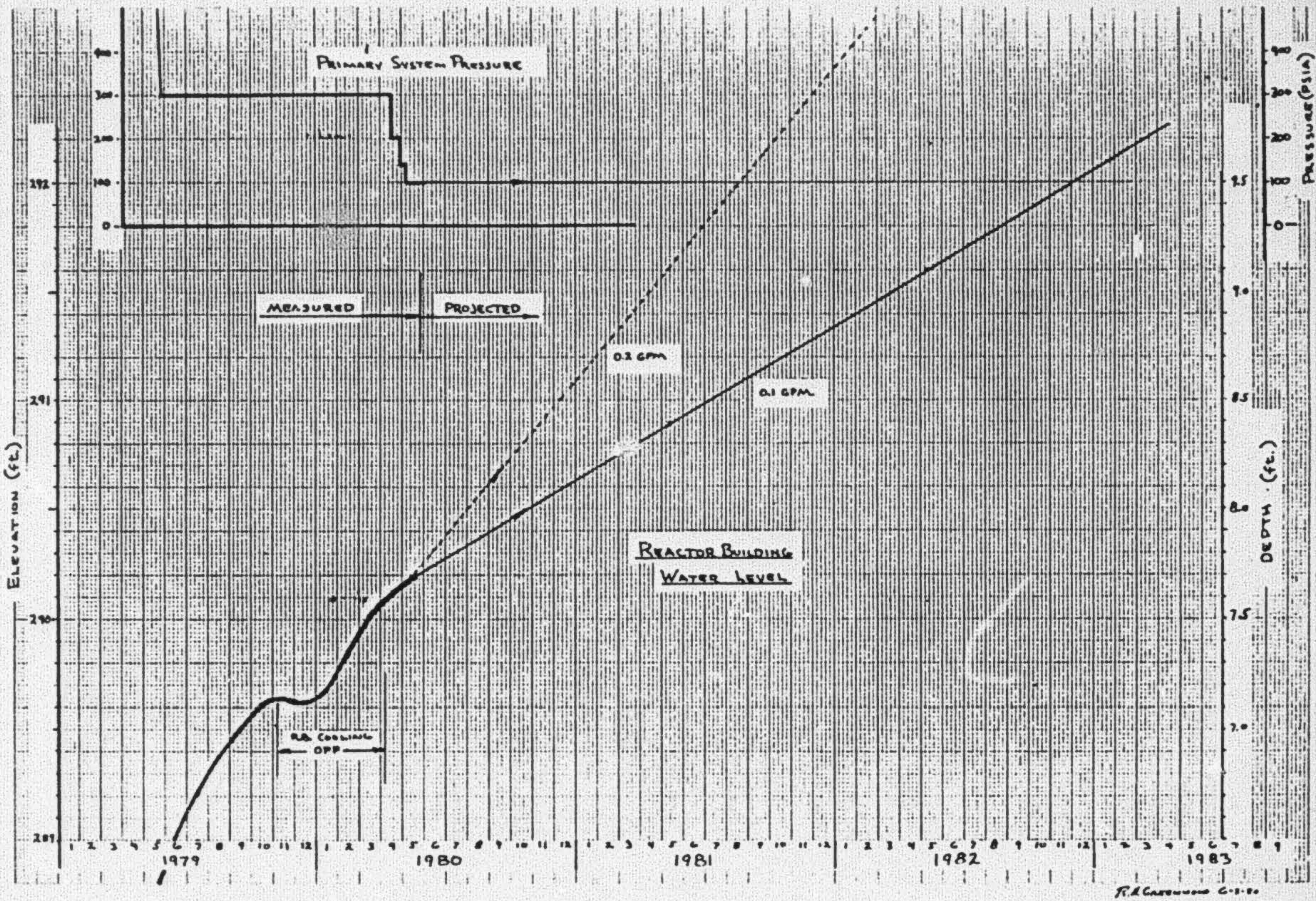
As a bounding (worst case) estimate - a higher leak rate was also studied. An inflow leakage of 0.2 gpm was indicated from sump level measurements during the time that the RCS was maintained at ∿ 300 psi. Thus, an upper elevation of the water depth could be 10.7 feet by July 1982 should the RCS pressure be returned to 300 psi. The projected depth at the postulated leakage rate of 0.2 gpm would be 293.2 feet.

A plot of the measured Reactor Building water level indicates that a fluctuation can be expected from deactivation of the Reactor Building Cooling Water. This type of action allows the R.B. humidity level to approach 100%. As condensation occurs on all components that are cooler than air, the sump water is transferred to higher elevation collection areas. When these collection areas become saturated or filled, the accumulation rate in the sump resumes at a rate matching the inflow leakage rate. When the RB cooling water is activated the collected water evaporates due to the lower humidity and is returned to the sump by condensing at the cooling coils which are drained into the sump. The type of water accumulation fluctuation shown by this figure is expected to occur seasonally, thus, the projected straight line increase of 0.1 gpm will fluctuate depending on RB Cooling System operation, but the average leakage from the RCS will be 0.1 gpm.

2.2 The equipment that is located within the projected water level band which is required to operate are redundant valves DHV1 or DHV171. These valves, when opened or operated will allow the RCS isolation boundaries to be moved to valve DHV3 which is located outside of the Reactor Building. The bottom of the operators for these valves is located at approximately 292.5 feet while the valve bodies are at 291.0 feet.

Containment penetration R-401 is currently being used to measure the RB water level. Water level above the 291.8 feet elevation will allow sump water to fill the penetration sleeve. The measurement system installed through this penetration will still function with proper sealing of the penetrating instrumentation probes.

A list of all electrical equipment whose operability would be affected by water up to elevation 293.2 is tabulated and is attached. The listed items are not considered critical for use in plant operation. the maintenance of the safe shutdown of the reactor. However, DH-V-1 and DH-V-171 are critical components necessary for SDS processing of reactor coolant.


PRIMARY SYSTEM PRESSURE
MEASURED
PROJECTED
0.2 GPM
0.1 GPM
REACTOR BUILDING
WATER LEVEL
R.B. COOLING
OFF
ELEVATION (ft.)
DEPTH (ft.)
PRESSURE (PSIA)
292
291
290
289
9.5
9.0
8.5
8.0
7.5
7.0
400
300
200
100
0
1979
1980
1981
1982
1983

LIMITING ELEVATION FOR EQUIPMENT OPERABILITY	EQUIPMENT DESCRIPTION	PENETRATION NUMBER	PENETRATION ELEVATION	EQUIPMENT ELEVATION
289'0"	RC Drain Tank Temp. WDL-TE-1200	R-604	295'0"	289'0"
	MU-V2B (Flow Diagram 2024)	R-502	299'0"	
	MU-V1B (Flow Diagram 2024)	R-509	322'0"	
	MU-V2A (Flow Diagram 2024)	R-502	299'0"	
289'2"	CF-V115 (Flow Diagram 2024)	R-509	322'0"	289'2"
289'3"	MU-V1A (Flow Diagram 2024)	R-502	299'0"	289'3"
289'4"	DH-V2 (Flow Diagram 2026)	R-509	322'0"	289'4"
290'6"	Letdown Cooling Outlet Temp. IC-7-TE1 and TE2	R-604	295'0"	290'6"
	CA-V4A (Flow Diagram 2414)	R-502	299'0"	290'6"
291'0"	RB-Penetration Temp. R-400 RB-TE-3273	R-400	291'0"	291'0"
	" " " R-402 RB-TE-3275	R-402		
	" " " R-405 RB-TE-3769	R-405		
	" " " R-406 RB-TE-3279	R-406		
	" " " R-407 RB-TE-3271	R-407		
	" " " R-601 RB-TE-3327	R-601		291'6"
	" " " R-603 RB-TE-3331	R-603		292'0"
	" " " R-607 RB-TE-3339	R-607		291'0"
	" " " R-610 RB-TE-3345	R-610		291'6"
	RC Drain Tank Low Level WDL-LS-1208	R-405		291'4"

LIMITING ELEVATION FOR EQUIPMENT OPERABILITY	EQUIPMENT DESCRIPTION	PENETRATION NUMBER	PENETRATION ELEVATION	EQUIPMENT ELEVATION
291'0"	CRD Outlet HDR Flow IC-10-DPT	R-607	291'0"	351'0"
	RC Temp Hot Leg RC-4A-TE 2	R-406		352'8"
	RC Temp Hot Leg RC-4B-TE 3	R-610		352'8"
	Pressurizer Relief Outlet Temp RC-10-TE 2, 3	R-603		354'8"
	" " " " RC-10-TE 1	R-603		359'8"
	Pressurizer Spray Line Temp RC-11-TE	R-603		354'8"
	Source Range Preampl. NI-1	R-406		310'0"
	Compensated Ionization Chamber Assy. NI-4	R-610		322'0"
	Power Range Detector Assy. NI-5	R-406		322'0"
	" " " " NI-8	R-610		322'0"
	RCP-1A, 2A Feeder	R-506		331'0"
	Pressurizer Heater Power	R-400		-
	" " "	R-402		-
	" " "	R-407		-
	RCP-1A, 1B, 2A, 2B Control for Lube Oil Pumps	R-405		-
	" " " "	R-607		-
	RPS & SFAS (Red)	R-406		-
	CRD Instr.	R-601		-
	" "	R-603		-

LIMITING ELEVATION FOR EQUIPMENT OPERABILITY	EQUIPMENT DESCRIPTION	PENETRATION NUMBER	PENETRATION ELEVATION	EQUIPMENT ELEVATION
291'0"	RC Drain Tank High Level WDL-LS-1206	R-405	291'0"	291'6"
291'0"	SG 1A & 1B Shell Temp. SP-2A & 2B TE 5	R-603		302'2"
	Pressurizer Inlet HDR Temp. RC-9-TE	R-603		309'6"
	Core Flooding Tank Pressure 1A &1B CF-1-PT 1, 2, 3,4	R-607		324'6"
	Core Flooding Tank 1A & 1B Level CF-2-LT 1, 2	R-607		324'6"
	RCP-1A, 2A, 1B, 2B Motor Eccentricity RC-VE-7889, 7890, 7894, 7895, 7899, 7900, 7904, 7905	R-607		328'0"
	RCP-1A, 2B, 1B, 2B Shaft Phase Reference RC-VE-7891, 7896, 7901, 7906	R-607		328'0"
	RCP-1A, 2A, 1B, 2B Velocity Sensor RC-VE-7887, 7888, 7892, 7893, 7897, 7898, 7901, 7903	R-607		331'0"
	RCP-1A, 2A Lower Brg. Clr. Low Flow NS-FS-4024, 4026	R-405		334'0"
	RCP-2A Lower Brg. Clr. Outlet Temp. NS-TE-1425	R-604		334'0"
	RCP-1A, 2A Motor Air Cooler Lo Flow NS-FS-3977, 3979	R-405		337'0"
	RCP-1A, 2A Upper Brg. Clr. Flow NS-FS-4028, 4030, 4032, 4034	R-405		343'0"
	Vib. & Loose Part S/G 1A UP Tubes YM-VE-7022	R-408		346'0"

LIMITING ELEVATION FOR EQUIPMENT OPERABILITY	EQUIPMENT DESCRIPTION	PENETRATION NUMBER	PENETRATION ELEVATION	EQUIPMENT ELEVATION
291'0"	Upper Vessel Shroud Vib. YM-VE-7020, 7021	R-610	291'0"	325'0"
291'6"	RPS & SFAS (Blue)	R-610	291'6"	-
292'2"	DH-V171 (Flow Diagram 2026)	R-500	299'0"	292'2"
292'5"	DH-V1 (Flow Diagram 2026)	R-500	299'0"	292'5"
292'6"	SV-V10A (Flow Diagram 2414)	R-506	322'0"	292'6"
↓	SV-V11A (Flow Diagram 2414)	R-506	322'0"	292'6"

SPECIFIC COMMENTS

QUESTION 3

You state that the radiation dose at the surface of the sump water measures approximately 120 R/hr. State expected radiation dose rate at the same elevation after removing all waste water in the reactor building.

ANSWER

The activity present in the solids in the basement of the reactor building will serve as the principle source when the water is removed. The dose rates have been calculated to be between 2 and 10 R/hr. on contact.

SPECIFIC COMMENTS

QUESTION 4

You state that the construction of a waste storage facility would exceed two years. Describe the waste storage facility considered and state estimated radiation doses from that facility.

RESPONSE

For purposes of this study, the facility evaluated included the capability to store the radioactive liquid water in a long-term storage facility. This long-term storage facility would include not only the water retention capability but that shielding necessary to permit unlimited occupancy at the storage area boundary.

Irrespective of whether the facility is designed as an above-ground facility with adequate building wall thickness for shielding or the facility is an underground facility, it is estimated that design, construction and placing the facility into operation would require at least two years expenditure of effort.

For this facility, a detailed design was not performed. Shielding requirements were not calculated. Therefore, radiation doses, as a result of the facility, are not available.

Recently, AGNS has performed a detailed analysis of the cost and schedule associated with the design, procurement, and construction of a single intermediate level liquid waste (ILLW) storage tank. This analysis was performed under contract to DOE; the results are reported in AGNS Report 35900-4.2-26, dated November, 1979, entitled "Engineering Evaluations of Waste Handling and Storage Facilities". This report is based on the ILLW tank design basis of a single 300,000 gallon tank manufactured of stainless steel enclosed in a stainless steel lined concrete vault. Applicable conclusions of the report are:

1. Engineering, procurement, and construction for a single ILLW tank is projected to require 33 months to complete.
2. Costs (in 1978 dollars) for a single ILLW tank is estimated to be $11,000,000.

QUESTION 4 (Cont'd.)

3. Average contained activity concentration in the tank is approximately 123 μCi/ml.

Using the results of the AGNS report for direct application to a TMI-II liquid waste storage facility, yields the following conclusions:

1. Costs (in 1980 dollars) for a single ILLW tank (escalated at 10% per year) would be approximately $13,500,000.
2. Assuming the requirement to construct and place into operation three ILLW tanks, projected costs would be $40,500,000. Supporting facilities (such as a transfer system) and tie-ins would impose additional costs, both monetary costs and schedule costs, on the construction of the facility.

As a result of the foregoing discussion, the construction of a long-term storage facility to retain the containment sump water and the reactor coolant is not deemed to be a viable alternative for the following reasons:

1. The schedule for placing the facility into operation is not commensurate with the goal of expeditious decontamination of the TMI-II facility.
2. Implementation of the long-term storage option does not provide for ultimate disposition of the contaminated wastes.
3. Implementation of this option does not appear to have a distinct cost-benefit.

SPECIFIC COMMENTS

QUESTION 5

Include in your evaluation a solidification system that would utilize a large liner for direct solidification of containment sump waste water.

RESPONSE

Alternative III is devoted to direct solidification and disposal of reactor containment building waters.

A 55-gallon drum contains approximately 7 cubic feet of solidified material. Should the LL-5100 cask be used (with a special liner), approximately 130 cubic feet of solidified material could be packaged.

The following discussion revises the analysis presented as Alternative III of the SDS Technical Evaluation Report based on use of the LL-5100 cask.

1. The construction of an on-site solidification facility would be required.
2. Based on 1,000,000 gallons of contaminated water to be processed, a 70 gallon availability of water volume in a 130 cubic feet liner, 70% availability of the solidification system, 24 hour/day operation, and a two-hour cycle time, the processing time may exceed 40 months.
3. Based on 1,000,000 gallons of contaminated water to be processed and a 70 gallon availability of water volume in a 130 cubic feet liner, the number of solidified liners that would be generated would exceed 14,285. Handling and transportation of this extremely large quantity of solidified waste would be prohibitive.
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

Because of the potential for a relatively high occupational dose commitment, in addition to the extremely large number of radwaste shipments, this alternative course of action is not deemed to be feasible. Furthermore, because of the long time for processing, this alternative is not considered a viable one in terms of expeditious clean-up of the TMI-II facility.

SPECIFIC COMMENTS

QUESTION 6

What is the "desired bed loading" for the first zeolite bed. Discuss the manner in which the loading will be determined during operation. Also, discuss loading in terms of integrated dose rate limitations and the potential for gas generation while in store.

RESPONSE

The desired bed loading on the first zeolite bed is that resulting from processing approximately 15,000 gallons. This yields an estimated deposition of 1590 curies Cs 134 and 9085 curies Cs 137 based on 10 cubic foot zeolite beds.

Bed loading will be calculated from the routine sample analyses results. Samples will be drawn from the Hi-Rad Feed and Intermediate Level Sample boxes, approximately once every four hours during system operation. Integration of influent radionuclide concentration minus effluent radionuclide concentration yields an accurate estimate of bed loading.

Zeolite is an inorganic material composed of a silicate of aluminum that poses no known limitation in terms of integrated dose or gas production.

SPECIFIC COMMENTS

QUESTION 7

Is it the intent to store the organic cation bed underwater in the storage rack?

RESPONSE

It is our intent to dispose of the organic cation beds in accordance with applicable regulations. However, underwater temporary storage is available and will be used as necessary.

SPECIFIC COMMENTS

QUESTION 8

You state that cation resin will be limited to less than 75 Ci of strontium based on zeolite bed effluent monitoring. State strontium analysis capability at the TMI site to sustain SDS process without delay.

RESPONSE

Radiochemical preparation areas will be installed in close proximity to the SDS operating area on the fuel handling floor. Reagents and apparatus necessary to perform strontium separations will be available for use. Following preparation for counting, samples will be analyzed for total strontium on gas flow or end window G-M equipment dedicated to support SDS operations.

This analytical plan will provide accurate strontium results in a minimum time. In addition, the data gathered during initial operation may be compared to on-line beta monitor readings to assure further control of cation bed strontium loading.

SPECIFIC COMMENTS

QUESTION 9

Describe design and operational features of beta monitors in detail. Are they designed for cesium or strontium breakthrough?

RESPONSE

The beta monitors are end-window G-M tubes manufactured by LND, Inc. tube number 7232. These tubes have a mica window of 1.5 - 2.0 mg/cm^2 thickness and operate at approximately 1000 volts. Five tubes are mounted in beta monitoring manifold to view zeolite bed effluents as well as cation influent and cation effluent sample lines. The tubes are mounted in holders, and shielded to reduce background. The holders are positioned to view thin wall sections of teflon tubing. The remaining monitors are mounted in line on the process input to the cation and the input to the leakage containment ion exchangers. They also view thin walled sections of process piping. Each detector is provided with an ASTON model 205 GM tube Pulse Discriminator to reduce false output pulse from low level noise. A cabinet will house the High Voltage power supply, Bertan Model 313A, liner ratemeters, Mech-Tronics Nuclear Model 1254, and recorder Tracor Westronics Model M11E. In addition, a radiation alarm, Ronan Model No. X15-10015 is provided.

The beta monitors are designed to provide assurance of proper bed function during initial operation and as a gross indicator of significant changes in process conditions. Correlation of beta monitor readings to analytical results from the sampling program will allow rapid estimates of process stream conditions so that SDS performance may be optimized and still assure that processed water is protected from recontamination in the event of an upset condition.

SPECIFIC COMMENTS

QUESTION 8

You state that cation resin will be limited to less than 75 Ci of strontium based on zeolite bed effluent monitoring. State strontium analysis capability at the TMI site to sustain SDS process without delay.

RESPONSE

Radiochemical preparation areas will be installed in close proximity to the SDS operating area on the fuel handling floor. Reagents and apparatus necessary to perform strontium separations will be available for use. Following preparation for counting, samples will be analyzed for total strontium on gas flow or end window G-M equipment dedicated to support SDS operations.

This analytical plan will provide accurate strontium results in a minimum time. In addition, the data gathered during initial operation may be compared to on-line beta monitor readings to assure further control of cation bed strontium loading.

SPECIFIC COMMENTS

QUESTION 10

At a loading of 75 curies of strontium on the cation bed, what is the expected cesium loading? What is the basis for limiting integrated does rate to 10^8 rads?

RESPONSE

Based on ORNL column tests, Cs 134 and Cs 137 will not deposit in any significant amount on the organic cation beds.

The integrated total dose of 10^8 rads for organic resin has been selected because radiation damage studies have indicated that degradation of resin may start to occur at higher total doses.

SPECIFIC COMMENTS

QUESTION 11

Provide estimates of the expected occupational exposures from the processing of sump and RCS water, and handling of all generated solid waste, including filters. Provide the basis for your estimates.

RESPONSE

The response to this question is currently being prepared. It will be submitted by July 31, 1980.

1-4

SPECIFIC COMMENTS

QUESTION 12

Identify those components which will require shielding, in addition to those in the fuel pool, to reduce dose rate to less than one (1) mR/hr. Also include components in the SDS offgas system.

RESPONSE

The SDS components above the spent fuel pool water level which are shielded are listed below. The piping associated with these components is also shielded as necessary.

1. Filter, Beta Monitoring, RCS, Ion Exchange, and Feedpump Manifolds.

2. High Rad Filter and High Rad Feed Sample boxes.

In addition, the polisher unit will be mounted in a CNSI 195-H cask (lid removed) to minimize radiation levels during processing.

The SDS offgas system is not shielded since only low radiation levels are expected. However, if required, temporary shielding will be added. The Mist Elimination tank and associated piping with its portion of the offgas system is mounted in the surge tank pot covered with a concrete slab (SEE DRAWING 527-A-5007).

SPECIFIC COMMENTS

QUESTION 13

Provide the basis for the 450 gallons water being released to the pool and for the 233,000 gallons of additional contaminated water.

RESPONSE

Paragraph 2.3 of the SDS Technical Evaluation Report provides a summary of the unexpected occurrences that are evaluated in Chapter 7, Accident Analysis. As presented in Chapter 7, 450 gallons of contaminated water is released from the final filter effluent line at a rate of 30 gpm for 15 minutes. The water leakage is released directly to the spent fuel pool, which has a capacity of 233,000 gallons. It is assumed that the leakage water mixes with, and is uniformly dispersed in, the spent fuel pool water.

SPECIFIC COMMENTS

QUESTION 14

You state, "It is possible that workers could be contaminated, however, prompt emergency decontamination procedures would prevent major radiation exposures". Elaborate this statement and describe the location and the method to be used for prompt emergency decontamination procedures. Provide the basis for the 11R/hr. maximum exposure rate assumed.

RESPONSE

Personnel decontamination that may occur in the unlikely event of this hypothetical accident, as with any other possible personnel contamination events, would be performed in accordance with the existing TMI-II procedures for personnel decontamination. These procedures have been developed in accordance with the requirements of the TMI-II Radiation Protection Plan. Personnel decontamination would take place in the personnel decontamination area located within the TMI-II Health Physics area.

The basis for a radiation level of 11 R/hr. has been provided in our letter of May 27, 1980 (TLL 251).

SPECIFIC COMMENTS

QUESTION 15

Discuss utlimate disposition of generated solid radwaste in greater detail. Describe storage capability for spent liners, spent resin processing, cask availability for shipment, and acceptability of the processed waste at a low-level waste burial facility.

RESPONSE

Storage capabilities for storage of spent vessels has been addressed in our response to Specific Comment #17.

Our present plans identify the use of the CNSI 1-13c casks for shipment of the zeolite and cation resins and the dewatered filters. Six of these casks are scheduled to be made available in January, 1981 for exclusive use by TMI for disposal of SDS generated wastes. The CNSI 195H cask will be used for shipment of dewatered polishing resins; two will be dedicated for exclusive TMI use.

Criteria which have been applied to all activities associated with the cleanup effort have included packaging of radioactive waste so that it can be transported to, and accepted by, waste disposal sites in compliance with existing regulations. We recognize the concerns which exist as to the form, content and utlimate disposal of the material collected by the proposed operation of the SDS. We are also aware of the various proposals to modify the existing regulations. Because of these circumstances, we have made provisions, in the design of the SDS and the availability of interim on-site storage for contaminated resins, to permit proceeding with the immobilization of the fission products without foreclosing future options for treatment to improve their suitability for long-term off-site storage or disposal. These features of our planning are key elements in our rationale for proceeding with the design, procurement, and installation of the SDS.

Your letter indicates "further guidance" will be available concerning solid waste disposal as work is pursued on the PEIS. We urge that resolution of these issues be given very high priority through an interagency task force with representation from all the concerned federal agencies.

SPECIFIC COMMENTS

QUESTION 16

You state that the results of recent experimental work with simulated and actual water samples from TMI-II were used to support the selection of specific ion exchanger materials for the SDS. Provide a copy of such test results to the NRC for review.

RESPONSE

The attachment to this response is extracted from a preliminary report generated by the Oak Ridge National Laboratory. This report is entitled "Post-Accident Cleanup of Radioactivity at the Three Mile Island Nuclear Power Station", ORNL/TM-7081. This attachment presents not only the report of specific ion exchanger selection tests, but the development of the SDS processing scheme. The final report will be available in August, 1980.

3.4 Treatment of High-Activity-Level Water

The largest volume of water generated from the accident can be regarded as high-activity-level water (HALW), which may be defined as water with a radionuclide concentration in excess of 100 μCi/ml based on the ^{137}Cs content.

3.4.1 Quantity and characteristics

Water in this category is primarily from two major sources. The first source (90,000 gal) is the water contained in the primary loop coolant circuit which is used to maintain the reactor in a safe condition by removing heat in a natural convection mode. The second source (540,000 gal) includes (1) the water contained in the reactor containment structure which resulted from the release of water from the primary cooling circuit during the early phase of the accident, (2) the volume of liquid transferred into the building through the containment spray system, and (3) a large volume of water which was released through miscellaneous equipment (pump seals, space coolers, etc.) during the post-accident period. Depending on the fission product removal efficiency of the Epicor-II treatment system, the water in the three reactor coolant bleed tanks may be processed later, if necessary, in this system. Table 2 outlines the volumes of solution and concentrations of the primary radionuclides contained in this material.

In addition to the increased radioactivity contained in this water, the ionic contamination mandates that the water be given special treatment. The major ionic constituent is boron, present at levels up to approximately 2600 ppm, which is used as a neutron poison in the coolant circuit. In addition to this element, sodium is also present as the result of spray activation with NaOH solution. Table 3 presents significant information related to the chemical and ionic characteristics of the HALW solution.

3.4.2 Necessity for emergency tankage

Soon after the accident, it was recognized that additional tankage would be required to receive the HALW accumulating in the Reactor Containment Building. There was considerable concern that the water level in the

Table 2. Volumes of solution and concentrations of principal nuclides contained in high-activity-level water and precipitate[a]

	Primary loop[b]	Reactor Containment Building[c] Top	Middle	Bottom
Volume, gal	90,000	←	540,000	→
Radionuclide content of water, μCi/ml				
^{137}Cs	90-110	176	179	174
^{134}Cs	18-22	40	40	40
^{90}Sr	17-19	2.7	2.9	2.8
^{89}Sr	305-330	43	41	42
^{129}I		0.079[d]	0.080[d]	0.076[d]
^{131}I	0.5-0.6	0.012	0.012	0.013
$^{140}Ba/^{140}La$	4-5	0.09	0.078	0.14
^{3}H		1.03	1.05	1.01
Gross α		3.4 ± 1.6[e]	1.2 ± 1.3[e]	5.4 ± 2.0[e]
Radionuclide content of precipitate, μCi/ml (on total ml of bottom sample)[f,g]				
^{95}Nb				0.13
^{95}Zr				0.05
^{103}Ru				0.06
^{129m}Te				0.40(±0.17)
^{131}I				0.013
^{134}Cs				0.015
^{137}Cs				0.064(±0.020)
$^{140}Ba/^{140}La$				0.04/0.11
$^{89+90}Sr$				2.78 (one sample only)

[a]Radionuclide content of surface deposit on painted steel plug, μCi:

^{95}Nb - 1.7	^{125}Sb - 0.5	^{131}I - 0.3
^{103}Ru - 0.6	^{125m}Te - 0.5	^{134}Cs - 0.5
^{106}Ru - 0.4	^{127m}Te - 7.8	^{137}Cs - 2.1
^{113}Sn - 0.2	^{129m}Te - 23.6	^{144}Ce - 0.2

[b]As of July 17, 1979.

[c]As of August 28, 1979. Samples from top and middle--light yellow solution; sample from bottom--greenish with dirty green precipitate. All samples have a similar radiation level; pH is 8.1 (as measured in the top sample).

[d]Units are μg/ml.

[e]Units are dpm/ml.

[f]Average of two samples.

[g]As of August 29, 1979.

Table 3. Chemical and ionic characteristics of high-activity-level waste solution and precipitate

Element	Containment vessel liquid (pool) Top	Middle	Bottom
	Sample concentration, μg/ml (ppm)		
Al	3	3	3
B	1950	2200	1900
Ca	10	10	8
Cl	10	15	8
Cs	0.6[a]	0.7[a]	0.7[a]
Cu	≤0.2	≤0.2	10
Fe	0.6	1.1	1.8
K	4	4	4
Li	1.6[b]	1.6[b]	1.4[b]
Na	1080	1200	1200
Ni	≤0.2	≤0.2	3
P	0.3	0.3	0.2
Rb	0.3	0.3	0.3
S	9	8	7
Zn	0.5	0.5	0.4
Pu	0.010[c]	0.011[c]	0.033[c]
U	7[c]	13[c]	28[c]
	Solids from bottom sample, μg/ml (ml of total bottom sample)		
In			0.16
Cu			54
Al			7
Ca			≤2

[a]Fission product.

[b]Li; >99% ^{7}Li.

[c]ppb, not μg/ml.

building might rise high enough to flood and thereby prevent operation of some of the vital instruments that monitor and control the reactor. These instruments had not been designed for operation in a hostile environment of high radiation fields and submergence in liquids. Subsequently, a tank farm was designed and installed in the fuel storage pool A contained in the Unit 2 Fuel Handling Building. The original scheduled time for this project was 90 days; however, some acceleration of the schedule was required when the leakage rate increased and it appeared that 50,000 gal of primary coolant might have to be transferred to one of the primary coolant bleed tanks. But, two major leaks were repaired, and this transfer of primary coolant was deemed unnecessary.

The design of the tank farm incorporated six tanks with a total capacity of 110,000 gal. Two 25,000-gal tanks were installed near the bottom of the pool and connected to each other with a standpipe for which devices were to be designed later for sampling and solution transfer. Figure 8 is a photograph of these tanks taken during installation in the fuel storage pool. Four 15,000-gal tanks, installed above the 25,000-gal vessels, were connected to a second standpipe. Shielding for all vessels was provided by installing concrete slabs on the top of the pool structure. An independent off-gas treatment system was also installed on top of the tanks to decontaminate any gaseous effluent that might be evolved during tank operations.

Fortunately, the tank farm system was not needed for its original purpose. This eliminated the potential risk of transferring additional liquid out of the Containment Building (into the tank farm) and, thereby, increasing the inventory of iodine in a structure external to the reactor system. The decision had been made not to disturb the solution in containment so the 8-day ^{131}I could decay to safe limits (see Sect. 3.4.3).

The tank farm will be utilized as an integral part of the HALW treatment system.

3.4.3 Processing justification

In order to proceed with the recovery of the reactor system, the liquid being held in the Containment Building will require removal and treatment. Currently, this liquid is standing at a depth of approximately 7-1/2 to 8 ft

PHOTO B-3925-40

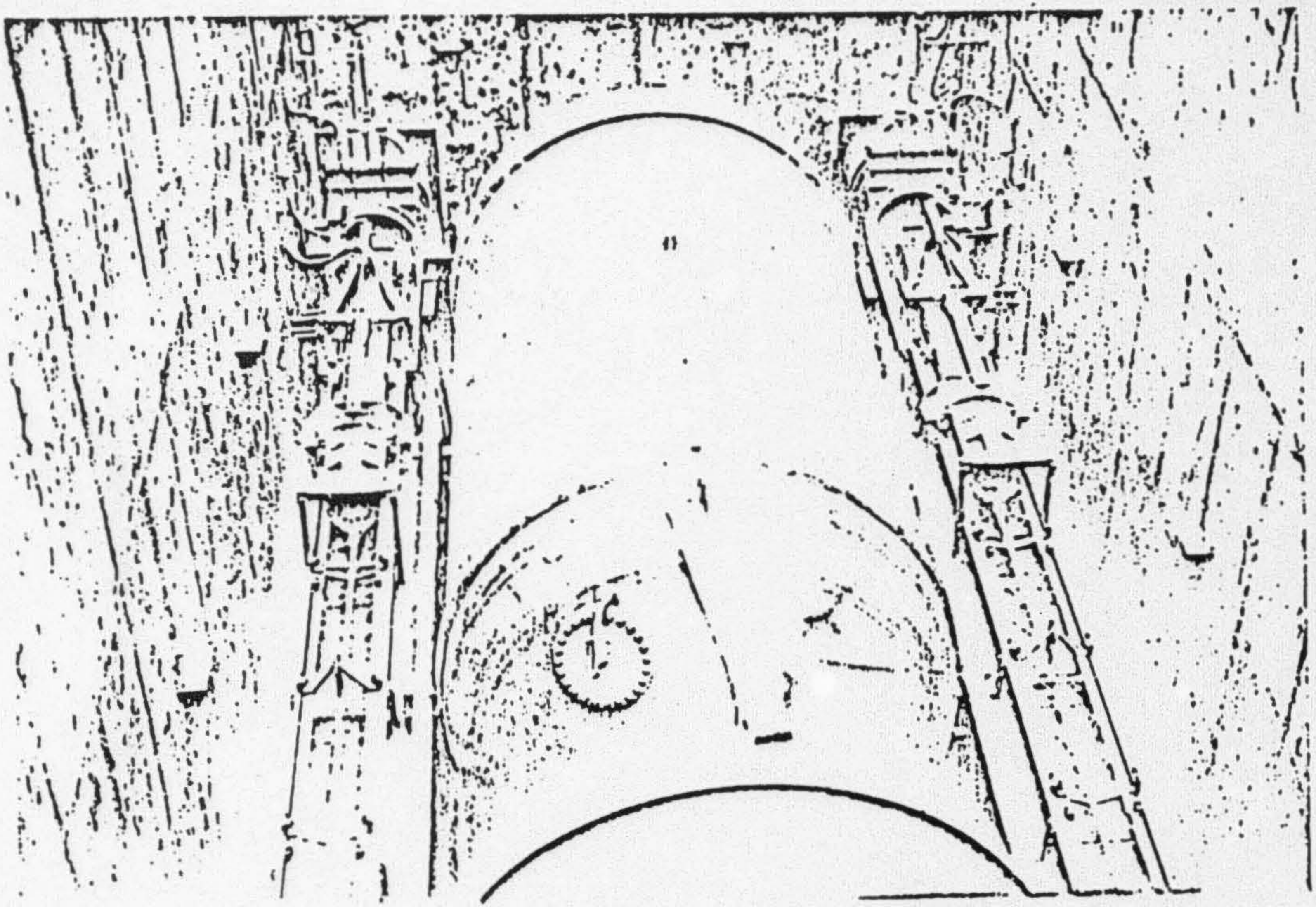


Fig. 8. Tank farm in the TMI-2 fuel storage pool during installation.

and is covering several components, including instruments. Although the leakage of water into the containment area has been minimized, the possibility for increased leakage continues to exist. In the early phases of the accident, the water in containment had ^{131}I concentrations estimated to be of the order of 10^4 μCi/ml. Because of this factor, every effort was made to avoid disturbing this solution until the radioiodine had been allowed to decay. Figure 9 shows the fission product decay curves, based on the analysis of primary coolant, for the radioactive nuclides that were of greatest concern with regard to treatment of the solution and indicates the most desirable processing period.

3.4.4 Flowsheet development

Because of the unique nature of the water to be decontaminated, certain phases of the development of a flowsheet were required. The first phase of the flowsheet development work involved the selection of a suitable exchange medium for the processing of this solution. Limited samples of actual TMI primary loop water were tested in an ORNL[2] hot cell to establish the characteristics of sorbents believed to be selective for the predominant $^{134-137}Cs$ and $^{89-90}Sr$ isotopes. A total of seven materials, both organic and inorganic ion exchange sorbents, were tested (Table 4). Results from these tests established that one of the zeolites, AW-500 (an inorganic exchanger), was highly selective for the cesium isotopes under the conditions prevailing at TMI and that Dow HCR-S, an organic resin, would be suitable for selection of the strontium isotopes. Following these studies, a series of small-scale column runs was made to verify the number of column volumes that could be expected from the various sorbents using solutions adjusted to TMI's water conditions. Information resulting from these studies was then applied to a "scale-up" computer program at the Savannah River Laboratory to provide information on design of the ion exchange columns. Pertinent information yielded by these studies is presented in Figs. 10 and 11.

3.4.5 Process design objectives and criteria[8]

Chem-Nuclear Systems, Incorporated (CNSI), was awarded a contract from the General Public Utilities Services Corporation to provide a flowsheet for

ORNL Dwg 79-13239A

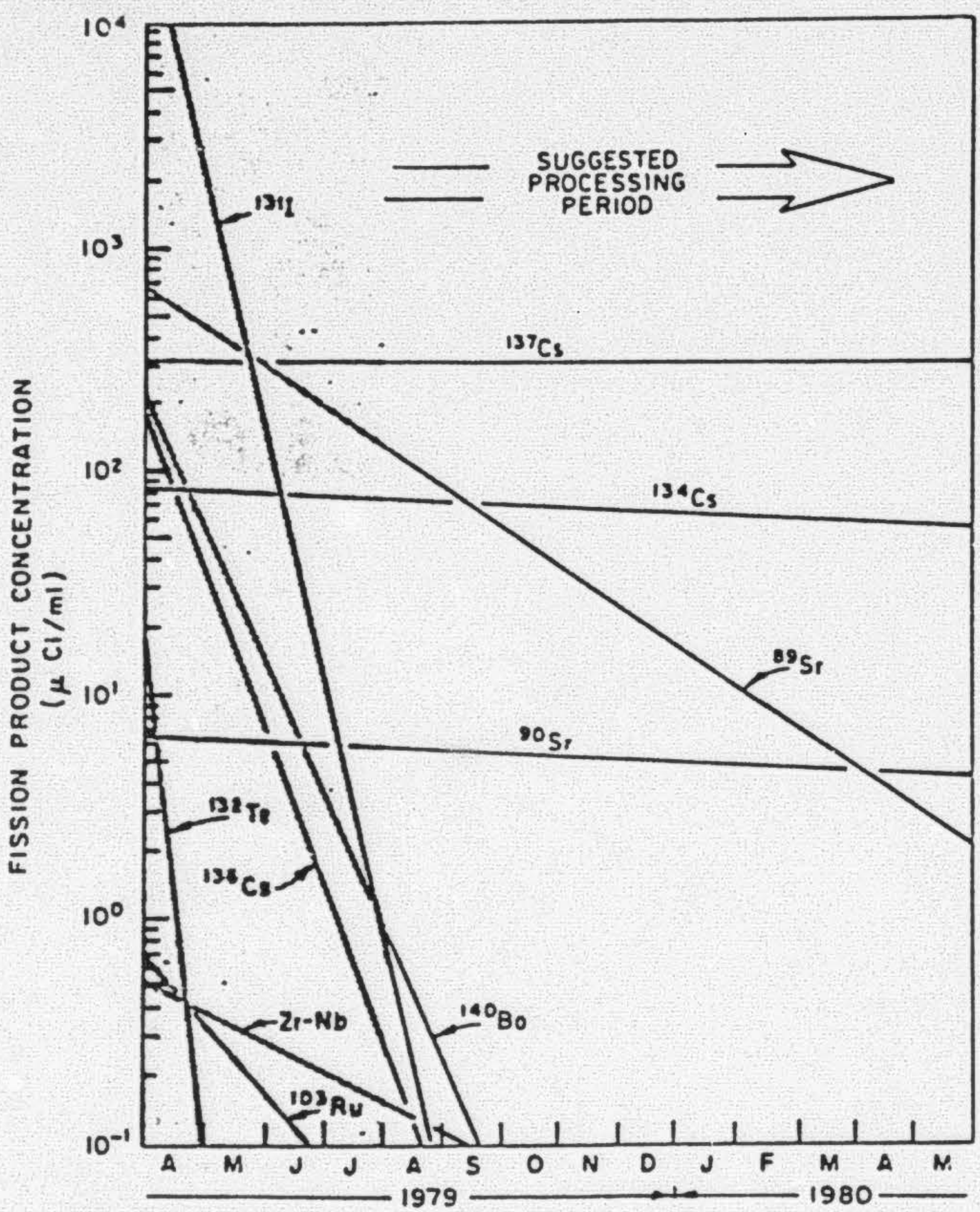


Fig. 9. Fission product decay in the TMI-2 primary loop.

ORNL Dwg 79-15328

Table 4. Distribution coefficients (K_d) for primary coolant

Exchanger	Sr K_d 5 min	Sr K_d 2 hr	Sr K_d Second pass	Cs K_d 5 min	Cs K_d 2 hr	Cs K_d Second pass	Final pH	Phase ratio (ml/g)
	Primary water as received – 960 ppm Na							
HCR-S	50	230	20	3700	8700	>330	3.6	27
IR-200	120	290	28	85	200	250	8.3	71
Z-900	~0	50	39	280	1800	1400	--	144
AW-500	~2	100	--	180	1300	--	--	136
Clino.	22	96	130	135	990	3000	8.6	67
Glass	96	250	54	160	440	400	8.6	68
Charcoal	20	50	60	0	0.7	2	8.7	42
	Primary water + 50% NaOH to give 4000 ppm Na							
HCR-S	0	265	19	40	48	9000	8.5/3.7	56
Z-900	~0	---	--	140	--	--	--	55
AW-500	27	54	86	238	1400	2200	9.7	54
Clino.	20	42	40	75	400	1300	10.1	60
Glass	55	270	38	70	350	320	10.1	56
Na Titanate	1500	4200	16	32	70	70	10.1	58

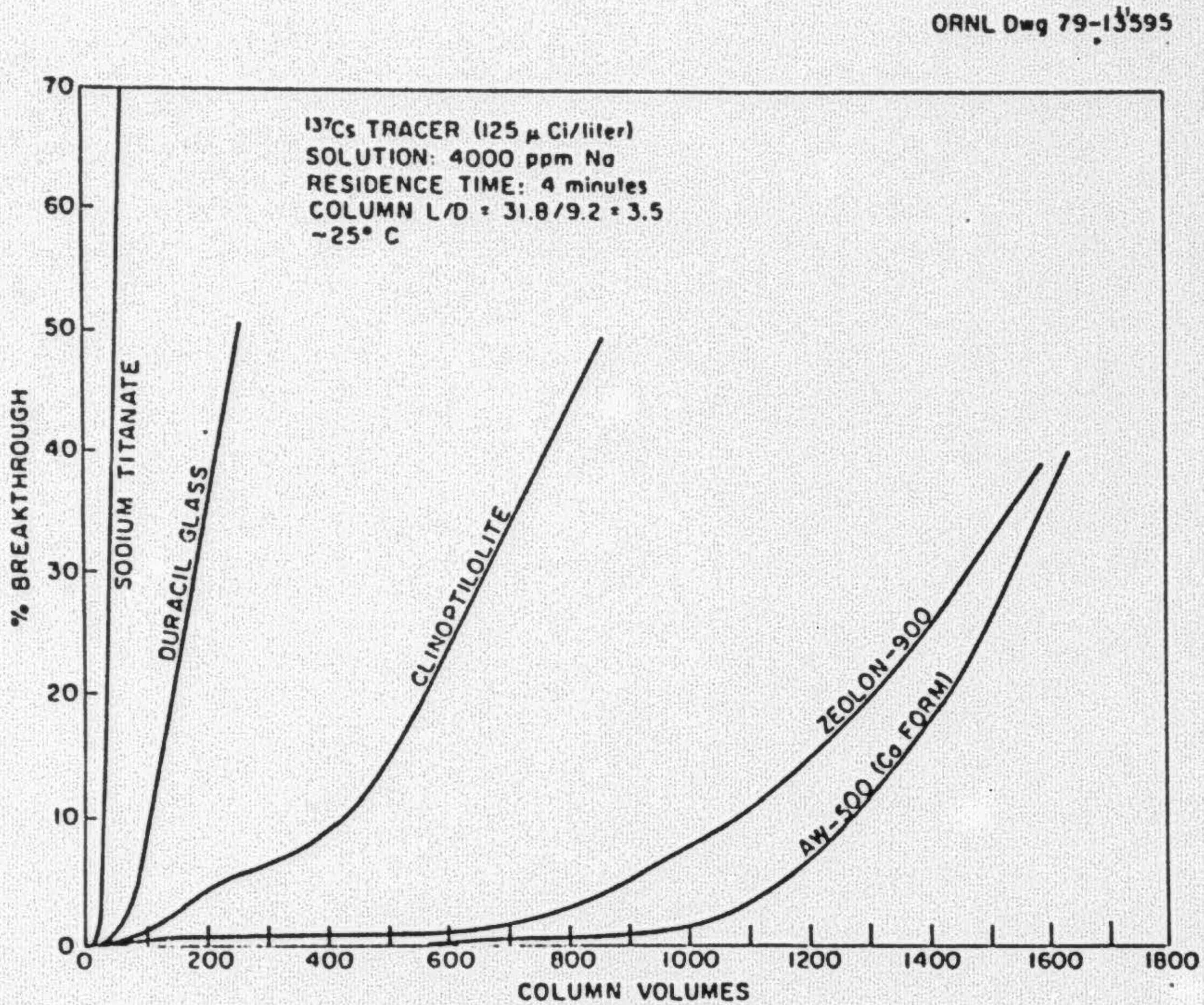


Fig. 10. Small-column loading tests of ion exchange materials with synthetic TMI-2 primary loop water and ^{137}Cs tracer.

ORNL Dwg 79-13594

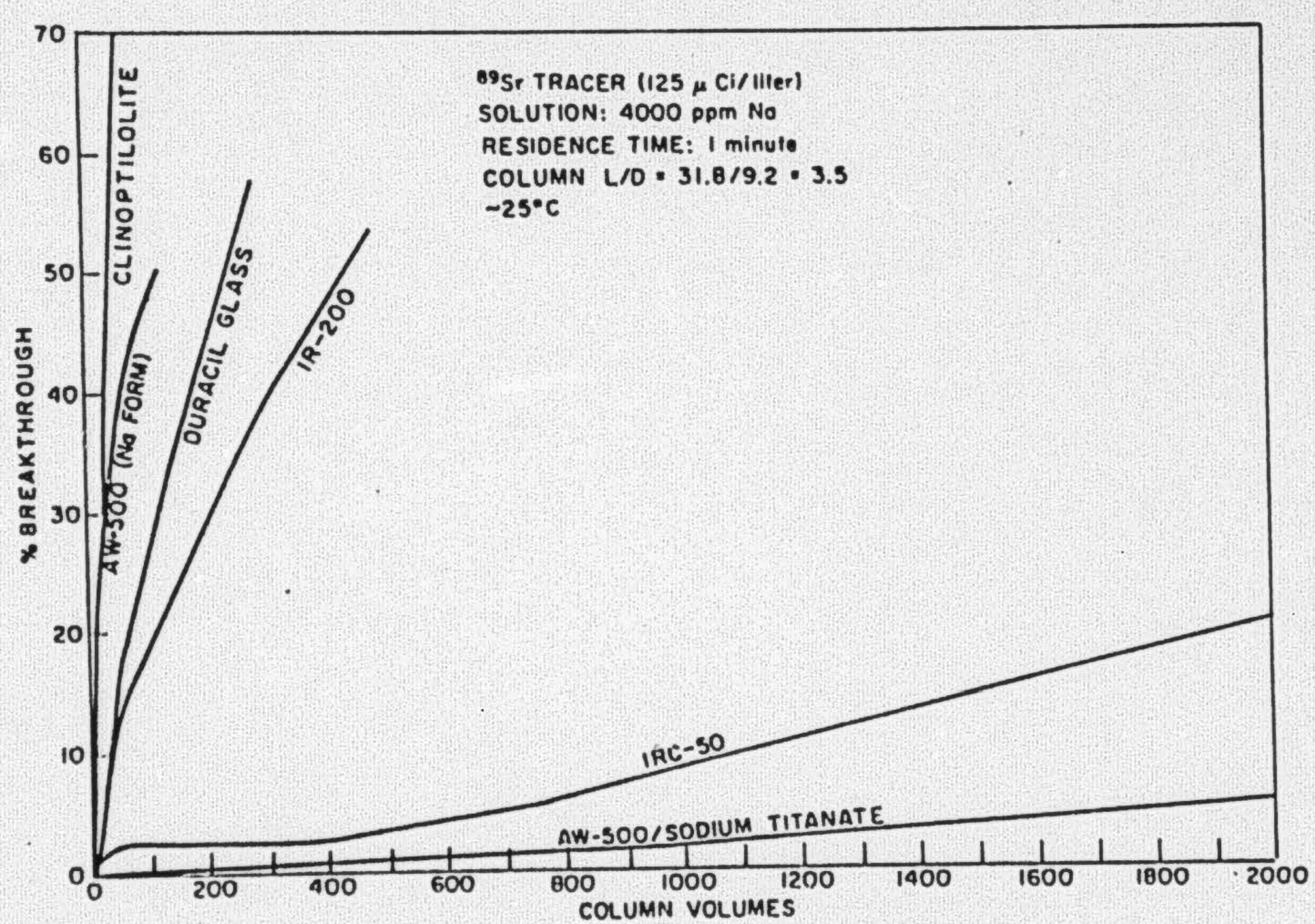


Fig. 11. Small-volume loading tests of ion exchange materials with synthetic TMI-2 primary loop water and ^{89}Sr tracer.

the treatment of HALW. CNSI then employed Allied-General Nuclear Services to assist them in the design of a suitable system.

A series of design objectives was drafted to provide a system that

1. was totally integrated and as independent as possible from existing waste and off-gas systems at TMI,
2. could remove the fission products in the water to a level that would meet designated TMI release requirements,
3. could be operated with a minimum of exposure to personnel and a negligible risk to the general public, and
4. would accomplish these objectives in a timely and cost-effective manner.

The following design concepts were used in order to accomplish these objectives:

1. CNSI has designated an ion exchange system for the removal of radionuclides from the water.
2. The ion exchange columns are to be located underwater in the spent fuel pool, which will provide for containment and personnel shielding.
3. To the extent possible, all-welded stainless steel construction is specified to minimize the potential for leakage.
4. Lead or equivalent shielding is provided for pipes, valves, and vessels (except those located underwater) that could carry contaminated water.
5. Contaminated water from the containment sump is to be filtered, collected in one of two batch tanks, sampled, and then pumped continuously to the ion exchange system.
6. Two parallel filters and ion exchange systems are to be used to provide capacity and to ensure continuous operation.
7. Continuous effluent monitoring systems are to be provided to ensure adequate control of the operation.
8. The valving systems are to be designed to minimize the impact of operator error.
9. The system is to be designed for a minimum of 2 years of service.

3.4.6 System and process description

The conceptual system design consists of the following elements:

1. a feed filtering system;
2. four feed tanks with independent mixing, sampling, level-measuring, and feed-pumping capability;
3. two parallel primary ion exchange trains, each comprised of three 7-ft^3 beds of zeolite exchange media;
4. two parallel ion exchange beds containing organic cation resin for removal of strontium;*
5. a continuous monitoring system and sampler system for control of bed loading;
6. a secondary containment system for the filters, primary and cation beds, and radiation shielding for piping, valves, samplers, and monitors;
7. a carbon-cation "filter" bed for removal of trace fission products;
8. a mixed-resin ion exchange polishing bed for removal of trace fission products that are not trapped on the primary or cation beds;
9. a 195-ft^3 ion exchange bed for removal of boron (should this be required);
10. two monitoring tanks for collecting and sampling the treated water prior to discharge and storage;
11. an off-gas system for treating and filtering gases and air from the system;
12. associated piping, valving, and structural supports required for placement of the system components; and
13. auxiliary systems, including underwater ion exchange column storage, a column dewatering system, a system for solidifying the resin beds with concrete, and analytical equipment.

Feed filtering system. This system consists of two parallel underwater filter units. Each unit is a cluster of 5-μ cartridge filters protected by perforated metal screens and contained in a 24-in.-diam x 54-in.-long column. Inlet, outlet, and vent connections are made with Hansen quick-release valved couplings which are remotely operated from the top

*About 90 to 95% of the strontium and essentially all of the cesium will be loaded on the zeolite.

of the pool. A gamma monitor, located in a dry well adjacent to the filters, and inlet and outlet pressure gauges are provided to monitor and control solids loading. Load limits will be based on available system pressure and/or the surface dose limit for the filter cask. The capacity of each filter is about 15 gpm. Feed will be pumped from the reactor containment sump, through the filters, and into the feed tanks on a batch basis.

Feed tank system. This system consists of four of the 15,000-gal tanks located in the emergency tank farm. Each tank will have an air-lift system for circulating the feed solution and for raising it to a small pump head tank (approximately 8 in. diam x 3 ft long). A sampler will be provided on each tank to permit characterization of the feed prior to sending it to the ion exchange system. Level-measurement instruments are also provided. Feed will be pumped to the ion exchange columns using one of two seal-less, magnetically coupled pumps rated at approximately 15 gpm each. The samplers, headpots, and pumps will be located in a shielded enclosure.

Primary ion exchange columns. This system consists of six underwater columns (24 in. diam x 54 in. long), each containing 7 ft^3 of Linde AW-500 zeolite resin. Inlet, outlet, and vent connections are made with remotely operated Hansen couplings (similar to the filter connections). The beds are arranged in two parallel trains of three each, with piping and valves provided to operate either train individually or both trains simultaneously. Loading will be controlled by feed batch size, operating time, effluent sample analyses, and continuous monitoring. Flowmeters are provided for the total feed stream and the individual feed streams to the two parallel trains. The maximum loading per column will be 65,000 Ci (cask limit). These columns are expected to remove greater than 99% of the cesium and 90 to 95% of the strontium from the water. When the desired bed loading is achieved on the first bed of the train, the flow to the train will be stopped. The treated water will then be displaced with fresh water (through the downstream beds), and the first bed will be disconnected and moved to the storage rack using the pool area crane. The second bed will be disconnected, moved to the first position, and reconnected. Likewise, the third bed will be moved to the second position, and finally a fresh bed will be placed in the third position. (*Note*: This approach is considered to be superior to a valving manifold which would permit rerouting of the streams because it minimizes the potential of a serious operator valving error. A valve switching approach was considered; however, a valving error could result in totally bypassing the primary beds. The current design prevents this from occurring.)

Cation columns. Two additional underwater columns (of the same size and type as the primary columns) are located immediately downstream. These columns are loaded with organic cation resin (probably Dow HCR-S) for removal of residual strontium. Column loading will be limited to 75 Ci of strontium based on primary column effluent monitoring and analysis. The columns are arranged to be operated singly or simultaneously in parallel. The curie limit is based on restricting the integrated radiation dose to the resin to less than 10^8 rads.

Sampling and monitoring system. As previously described, the feed will be collected in batches, sampled and analyzed, and fed continuously to the ion exchange system. Sampling lines are provided on the effluent streams of each of the primary beds and cation beds. A continuous sample flow from each operating bed effluent will be passed through a beta scintillation monitor. A gamma monitor is located on the primary effluent line, and a local area radiation monitor is located near the valve box. Each of the monitors will have an alarm system and a high-radiation trip point which will close an automatic valve of the main feed line, thereby stopping the operation in case of a leak or a serious breakthrough. Shielded samplers are also provided to permit sampling at any time. Continuous sample effluent is routed to the off-gas separator tank.

Containment and shielding. The filters, primary beds, cation beds, and their associated remote connectors and flexible lines are contained in submerged boxes designed to confine any contaminated leakage. Each box is connected to a pump manifold, and a continuous flow of about 10 gpm is maintained through each box. The containment pump effluent is discharged to an additional mixed-bed resin column and is released back into the pool. A large, shielded box is provided for the various valves and piping which are required for feeding, flushing, and venting the columns. This box will be located on the edge of the pool, with shielded lines entering and exiting the box from the bottom. Valves will be operated manually using extension handles that penetrate the top of the box. The valves will also be serviced from the top of the box. A separate shielded box will be provided for the column samplers. All lines which are normally radioactive, or could potentially be radioactive, are to be shielded. Shielding will be designed to limit the operator dose to 1 mrem/hr at contact. Wherever possible, pipe runs will be located underwater to minimize the need for lead shielding.

"Filter" bed. This is a 10-ft^3 ion exchange bed containing carbon and cation resin. This bed is intended to remove trace fission products from the water. Load control will be based on the cation effluent monitor and the column radiation level.

Mixed-resin polishing system. This system consists of a 195-ft^3 bed of mixed anion and cation resins enclosed in a special shielded cask. This unit, which is also intended to remove trace fission products from the cation effluent, will be located above the water. Load control will be based on the cation effluent analysis and the radiation level at the surface of the container.

Boron removal bed. If desired, a 195-ft^3 resin bed, enclosed in a shielded cask, is provided for boron removal. Effluent sample analysis and surface radiation level monitoring will be used to control loading.

Monitoring tank system. Two monitoring tanks of about 20,000-gal capacity will be provided for collecting and sampling the treated water prior to disposal. The tanks will be provided with mixing, level-measuring, and sampling systems, in addition to transfer pumps. Piping will also be provided for recycle of the treated water in the event that it does not meet disposal specifications.

Off-gas system. The feed tanks, valve boxes, monitoring tanks, and column vents are connected to this system. The system consists of a separator tank (which also serves as a sampler effluent collector and valve-box drain tank), an electric off-gas heater, a roughing filter, a charcoal bed for iodine retention, two absolute (HEPA) filters, and a centrifugal blower. The discharge of this blower will be monitored and routed to the reactor vent system. A pressure control system is provided for the blower to position, and finally a fresh bed will be placed in the third position. (*Note*: This approach is considered to be superior to a valving manifold which would permit rerouting of the streams because it minimizes the potential of a serious operator valving error. A valve switching approach was considered; however, a valving error could result in totally bypassing the primary beds. The current design prevents this from occurring.)

Cation columns. Two additional underwater columns (of the same size and type as the primary columns) are located immediately downstream. These columns are loaded with organic cation resin (probably Dow HCR-S) for removal of residual strontium. Column loading will be limited to 75 Ci of strontium based on primary column effluent monitoring and analysis. The columns are arranged to be operated singly or simultaneously in parallel. The curie limit is based on restricting the integrated radiation dose to the resin to less than 10^8 rads.

Sampling and monitoring system. As previously described, the feed will be collected in batches, sampled and analyzed, and fed continuously to the ion exchange system. Sampling lines are provided on the effluent streams of each of the primary beds and cation beds. A continuous sample flow from each operating bed effluent will be passed through a beta scintillation monitor. A gamma monitor is located on the primary effluent line, and a local area radiation monitor is located near the valve box. Each of the monitors will have an alarm system and a high-radiation trip point which will close an automatic valve of the main feed line, thereby stopping the operation in case of a leak or a serious breakthrough. Shielded samplers are also provided to permit sampling at any time. Continuous sample effluent is routed to the off-gas separator tank.

Containment and shielding. The filters, primary beds, cation beds, and their associated remote connectors and flexible lines are contained in submerged boxes designed to confine any contaminated leakage. Each box is connected to a pump manifold, and a continuous flow of about 10 gpm is maintained through each box. The containment pump effluent is discharged to an additional mixed-bed resin column and is released back into the pool. A large, shielded box is provided for the various valves and piping which are required for feeding, flushing, and venting the columns. This box will regulate vent system pressure. The separator tank, which will be located in the surge pit and covered with a concrete shield, is equipped with an automatic-level-controlled pump routed back to one of the feed tanks.

Piping and equipment arrangement. The feed pumps, feed samplers, and head pots will be located in shielded enclosures directly over the tank farm area. As previously described, the filters, primary beds, and cation beds are to be located underwater in special containment boxes. These boxes and columns are to be supported along one side of the pool on a structural steel rack which hangs from the edge of the pool. This rack, which also serves to support the system, will include an operating platform from which the remote connectors are to be handled.

Auxiliary systems. A storage rack, sized to handle the projected number of filters and ion exchange beds required to treat the containment sump water, will be provided. A special rack will serve to displace the water from the columns and filters and to dry them prior to loading into the transfer cask. This rack will be located in the cask loading pool. A system for solidifying the filters and ion exchange columns with concrete will also be provided. An additional system will be included for analyzing feed and effluent samples. Gamma and beta proportional counters and the necessary equipment for boron and sodium analysis will be provided as a minimum.

Figures 12-15 show details of the liquid waste (HALW) decontamination system. (The HALW treatment system is in no way related to either Epicor-I or Epicor-II.)

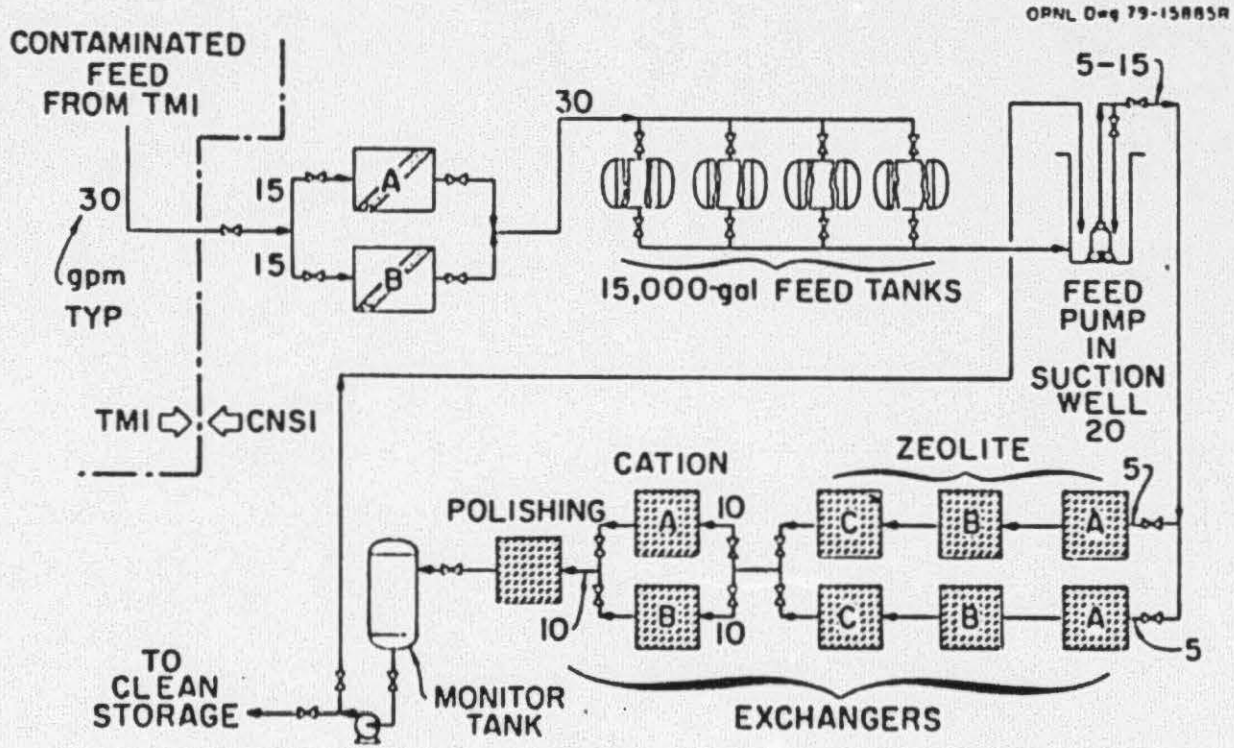


Fig. 12. Liquid waste decontamination flowsheet.

ORNL Dwg 79-15887R1

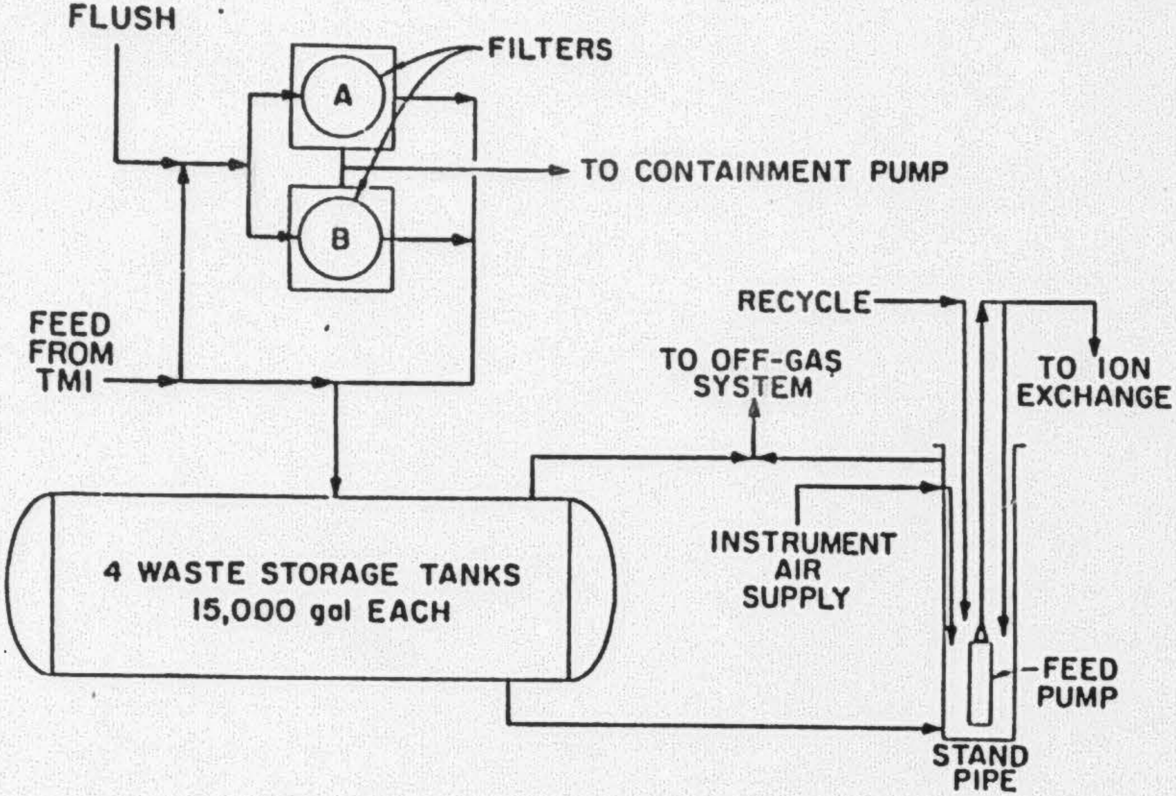


Fig. 13. Feed system details.

ORNL Dwg 79-15884R

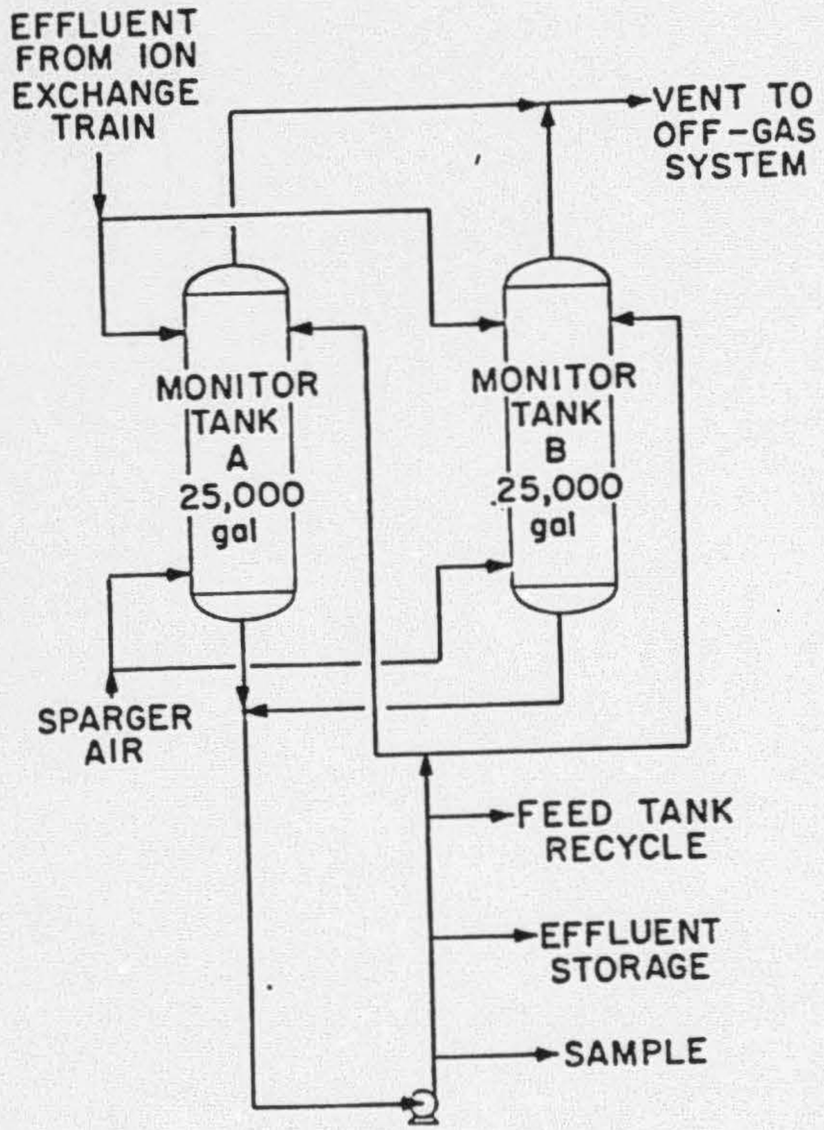


Fig. 14. Monitor tank details.

SPECIFIC COMMENTS

QUESTION 17

You state that only 200 bed volumes (approximately 2000 cubic feet or 15,000 gallons) of waste water are anticipated to be processed before the first zeolite bed is removed and replaced.

This will require more than 60 zeolite bed replacements (excluding the subsequent zeolite beds, cation beds, and filters) to process one million gallons of waste water through the SDS. Justify the storage capacity provided for only 40 spent vessels in the fuel pool.

RESPONSE

At the time the SDS Technical Evaluation Report was written, system design required the construction of facilities within the fuel pool or sufficient size to provide for the underwater storage of 40 spent vessels in the pool.

Upon receipt of further guidance, as indicated in the NRC letter dated May 28, 1980, H. R. Denton to R. C. Arnold, we intend to process SDS resins such that resin disposition options are not foreclosed. These options are: 1) ship dewatered SDS resins, 2) transfer SDS resins for solidification, or 3) transfer the SDS resins to interim on-site dry storage.

SPECIFIC COMMENTS

QUESTION 18

Provide the flow capacity of the pool secondary containment demineralizers.

RESPONSE

The secondary containment demineralizers have a flow capacity of 50 gpm.

SPECIFIC COMMENTS

QUESTION 19

Provide the administrative surface dose rate limit for change out of the SDS filters.

RESPONSE

Calculations from the ANISN computer code have indicated that contact surface exposure rates of 5000 R/hr on contact from Cs 134 and Cs 137 loaded on zeolite result in readings acceptable for shipment. In view of the fact that the majority of solids are Sr 89/90 beta emitters, surface exposure rates on filters should not become prohibitive before the maximum loading is reached. However, underwater monitors are located in close proximity to the roughing and final filters within the leakage containments. Administrative limits will be placed at a fraction of the above exposure rate to assure that spent filter vessels may be safely loaded in the CNSI 1-13c cask.

SPECIFIC COMMENTS

QUESTION 20

Describe how the shipping cask and its contained liner will be drained prior to movement out of the pool.

RESPONSE

The CNSI 1-13C cask is designed for underwater loading of vessels. With the drain plug removed and the vent open, the cask will be lifted above the pool surface and allowed to gravity drain.

The contained liner will be dewatered at the SDS dewatering facility prior to insertion in the shipping cask.

SPECIFIC COMMENTS

QUESTION 21

Describe how spent SDS filters will be changed out and packaged for ultimate disposal.

RESPONSE

SDS prefilters and final filters are packaged in vessels which are similar in design to the ion exchange vessels. These vessels are of the identical physical size and utilize Hanson quick disconnect fittings which allows the change out of vessels to be accomplished through the use of remote underwater handling tools similar to the tools used for the ion exchange vessels. Once the filters are changed out, they will be dewatered in the dewatering station, (as depicted on CNSI drawing 527D-A-5011). After dewatering, the filter vessels will be placed in the shipping cask (CNSI 1-13C) underwater. The cask will then be decontaminated and prepared for shipment via truck to the disposal site.

Additional studies of filter media and techniques are currently under investigation which could offer additional options for processing.

The present technique is in compliance with existing Federal Regulations and the ability to adequately perform the dewatering task has been studied and testing accomplished, with the results published in the attached AGNS document dated March 14, 1980, entitled, "Dewatering and Drying Tests of the CNSI-TMI Prefilter and Final Filter".

DEWATERING AND DRYING TESTS OF THE CNSI-TMI
PREFILTER AND FINAL FILTER

by

James D. Logue

March 14, 1980

Allied-General Nuclear Services
Post Office Box 847
Barnwell, South Carolina 29812

Prepared for
Chem-Nuclear Services, Inc.
TMI-II Recovery Project

ACKNOWLEDGMENT

The author gratefully acknowledges the assistance of Terry D. Bryson and Preston Fields of the Separations Production Department in the performance of these tests.

ABSTRACT

The dewatering and drying tests were conducted on the Submerged Demineralization System (SDS) to determine if they could be successfully dried. Dewatering and drying is required in order to meet existing and future criteria for shipment and burial of vessels. The tests were started on February 8, 1980, and concluded on February 29, 1980.

This report contains an evaluation of test results and recommendations needed to satisfactorily meet drying criteria.

TABLE OF CONTENTS

TABLES:

FIGURES:

1.0 INTRODUCTION

Commercial low-level waste burial site criteria requires that wastes be free of liquids; that is for wastes received until January 1, 1981, the amount of liquid allowed is less than 1% of the total volume, not to exceed one gallon and 0.5% of the total volume for liquid wastes received after that date. Federal Transportation Regulations [49 CFR 173.393(G)] require enough absorbent material to absorb at least twice the volume of liquids present in each package. The present packaging design for spent Submerged Demineralization System (SDS) prefilter and final filter vessels does not allow for the addition of absorbent material; therefore, the SDS filter vessels must be dry prior to shipment.

The purpose of these tests was to verify that the SDS prefilter and final filter vessels could be successfully dried; and that after drying, the SDS filter vessels could be stored for short periods of time under water without any in-leakage of pool water. These tests also determined if any weight must be added to the dry submerged filter vessels to prevent them from floating.

For these tests, each filter vessel was weighed empty and again after filling with water. To determine the length of time required to achieve dryness, the filters were blown down with dry air from 10 minutes to 24 hours. After each blowdown, any water which had not been removed from the filter vessel or that had drained back into the filter from the connecting piping was measured. After drying, the filter vessel was pressurized with air and stored in the pool to determine if there was any in-leakage of pool water.

2.0 CONCLUSIONS AND RECOMMENDATIONS

2.1 Conclusions

(1) The filters can be successfully dried and once dried can be stored for short periods of time under pressure without any in-leakage of water.

(2) The dried filters require the addition of ballast to prevent them from becoming buoyant.

2.2 Recommendations

(1) The specific ballast requirements for each type of filter will be calculated and incorporated in the design of the units.

(2) To prevent in-leakage of pool water, the male hansen connector on the prefilter inlet nozzle should be connected to its female counterpart before attempting dewatering and drying.

3.0 DESCRIPTION OF TEST

Two mock-up filter vessels were fabricated per CNSI Drawings (527-D-5010, Ion Exchange Mock-Up; 527-D-5002, Prefilter-125 Micron; 527-D-5003, Final Filter-10 Micron; and 527-D-5004, Prefilter/Final Filter Details), except that instead of using dished heads on the filter top head, flat plate was used and a pipe plug was installed in the bottom of the filter housings. Plexiglas was used for the vessel walls. The flat plate was used to reduce cost and fabrication times, the plug was installed to allow for posttest draining, and the Plexiglas was used to allow inspection of the filter internals (see Figures 3 and 4).

The volumetric capacity of each filter was determined by measuring the water needed to fill the filter to overflow. A weigh tank and scale was used for this measurement. After filling, the filter inlet nozzle was connected to a dry air supply and the filter blowdown nozzle was routed to the weigh tank. Dried compressed air was used for the blowdown operation. The air was regulated to 40 psig and 10 scfm; the pressure was limited by the Plexiglas.

After the initial amount of water was discharged, two blowdowns each at 10 minutes, 2 hours, 4 hours, 8 hours, and 24 hours were performed. After each blowdown, the vessel was visually inspected for wetness and the plug was removed to collect and measure the remaining water. To simulate actual conditions, the filter vessel was tested at approximately 20 feet below the weigh tank. A blowdown was also accomplished underwater to simulate remote conditions. (Figure No. 1 shows details of setup.)

After the filter unit was dried, lead was added for ballast. The extra weight enabled the filter to remain submerged after drying and added sufficient stability to the filter vessel to prevent it from shifting during hookup operations. The filter units were pressurized and submerged under approximately 20 feet of water and stored for 48 hours. After storage, the filters were removed from the pool and inspected for water.

4.0 TEST RESULTS AND EVALUATIONS

Following blowdowns of 10 minutes through 8 hours, the surface of the filters and the top and bottom plates remained wet. Liquid drained from the filters when the plug was removed. To achieve the dryness necessary to ship the filter vessels, blowdown times of 16 hours for the prefilter and 24 hours for the final filter are required. After the 16- and 24-hour blowdowns, the filter internals were dry and no liquid was detected when the drain plug was removed (see Table 2).

Dry compressed air at a temperature of approximately 68°F was used for these tests. The air was regulated to 40 psig and 10 scfm. Heated air and air at higher pressure and volume would enhance the drying. Increasing the pressure and heating the air could not be accomplished during these tests due to the pressure limitations of the Plexiglas.

The empty weight of the prefilter is 522 pounds and the final filter is 678 pounds. Extra ballast was required to submerge both filter vessels; in fact, 381 pounds of lead were added to the bottom skirt of each vessel. This additional weight allowed the vessel to be submerged and provided sufficient weight to prevent the vessels from shifting during hookup operations (see Table 1).

During the test, it was found that the male hansen connectors would open when submerged to a depth of 10 feet or more allowing in-leakage of pool water. After drying, the filters were pressurized to 15 psig and stored in the pool submerged under 20 feet of water. The filters were removed after 48 hours and no water was detected (see Table 1). Also, the check valves installed in the feed line of the prefilter will not hold pressure in the vessel and will allow in-leakage of water unless the plugged female hansen connector is installed.

5.0 ATTACHMENTS

Attachment 1 - CNSI-TP-2, Mock-Up Filter Ballast and Drying Test

Attachment 2 - Additional Instructions for CNSI-TP-02, Mock-Up Filter Ballast and Drying Test.

NO	REV	DATE OF THIS ISSUE	PREPARED BY	PAGE
CNSI-TP-02	0	DATE OF PREV. ISSUE	J. Logue	of

TITLE

MOCK-UP FILTER BALLAST AND DRYING TEST

1.0 INTRODUCTION

Contiminated water at TMI will pass through a set of roughing and final filters, before final purification in the ion exchangers. The filter units, after loading up, will be blown dry and shipped for burial. A mock-up filter unit will be used to simulate the drying process. The mock-up units will be filled with demin water and then blown dry. Data collected from this test will determine the amount of time and air to accomplish drying.

2.0 PURPOSE

2.1 This procedure provides the instructions to collect the data necessary to determine the time and amount of air required to remove enough water, so that the filters are considered dry and ready for shipment. Additional testing may be required.

2.2 Acceptance Criteria

2.2.1 This procedure will be complete when all steps are signed off; and

2.2.2 An evaluation is performed to determine if drying is sufficient.

3.0 SCOPE

3.1 Equipment mockup prefilter and final filter.

3.2 Area

FRSS

4.0 PREREQUISITES

Pressure test of mockup filters must be performed before units are used. Pressure test complete CYO L 2/8/80.
Cognizant Engineer T/D.

5.0 SAFETY PRECAUTIONS

5.1 Follows instructions as outlined in the BNFP Industrial Safety Manual.

5.2 DO NOT EXCEED 40 PSI on filter unit.

REVIEWED BY		APPROVED BY	
		[signature]	2/12/80
		[signature]	2/11/80

FORM OD-1

6.0 EMERGENCY ACTION

None anticipated.

7.0 TEST EQUIPMENT

7.1 350 Kg Scale

7.2 55 Gallon Drum

7.3 Humidity Testor

7.4 Dry Ice or Liquid Nitrogen

8.0 PROCEDURE

8.1 CALIBRATE 350 Kg scales per S-COP-90-3.

Operator TB T/D ____

8.2 Weigh Mockup filter units.

8.2.1 WEIGH empty units.

Prefilter 736.8 Kgs Final Filter 307.49 Kgs Operator TB

8.2.2 SUBMERGE filter unit into east CUP onto TMI test stand.

8.2.3 NOTE any difficulty to submerge.

Remarks Added 381 lbs of lead to both filters To submerge. Operator TB

8.2.4 REMOVE test unit from CUP.

8.2.5 FILL UNIT's with water.

8.2.5.1 ATTACH water hose to fill line.

8.2.5.2 ATTACH hose to vent.

8.2.5.3 FILL unit with water until it comes out of vent and DETACH hose.

8.2.6 WEIGH filled units. (weight of water only)

Prefilter 274.3 Kgs Final Filter 245.1 Kgs Operator TB

8.3 SUBMERGE filled filter unit into east CUP onto TMI test stand.

8.4 CONNECT regulated instrument air supply to addition nozzle and ROUTE discharge line from blowdown nozzle to drum and secure.

8.5 START air to filter unit at ~~2~~ 5 SCFM. (DO NOT EXCEED 40 PSI ΔP.
gol 2/1/80

8.6 SHUT OFF air ~~after 5 minutes~~. 10 mins after water stops coming out of hose.

8.7 WEIGH water in drum.

Prefilter 274.77 Kg Final Filter 242.05 Kg Operator gol

8.8 ~~REPEAT step 8.5 and measure humidity of discharge air.~~ Omit gol 2/8/80

~~Prefilter ____ Final Filter ____ Operator ____~~

8.9 ~~SHUT OFF air after 5 minutes and leave 20 to 30 psig pressure in filter unit.~~

~~Prefilter ____ psig Final Filter ____ psig Operator ____~~

8.10 REMOVE hose from nozzles and remove filter test unit from CUP.

8.11 INSPECT for water.

8.11.1 VISUALLY INSPECT internals for water.

Remarks Filters & internals wet ______ 2-9-80
Operator T/D

8.11.2 REMOVE plug from test unit bottom and measure any liquid present.

Amount 1350 ml ______ 2-9-80
Operator T/D

8.11.3 If unit appears to be dry, SET test unit aside for 48 hours and repeat 8.11.2.

8.11.4 If unit contains water contact Cognizant Engineer for further instructions.

Operator ______ T/D 2/9/80

8.12 EMPTY drum per instructions of Shift Supervisor. See AWR 30091 14416 for further Testing gol 2/8/80

8.13 REPEAT steps 8.2 thru 8.12 for final filter.

NO	REV	DATE	PAGE
S-COP-90-3	1	5/2/75	20 of 24

350 Kg Scale

Scale Calibration Data Sheet

for Vessel TMI-Tool* Pass No. ________*

Weight* Transfers	Net on Scale-lb.	Net on Scale-Kg	Scale Reading	**Difference
0	0	0	0.00	0.00
+ 5	5	2.263	2.27	+.002
+10	15	6.804	6.81	+.006
+10	25	11.340	11.36	+.020
+50	75	34.019	34.04	-.021
+50	125	56.699	56.72	+.021
+50	175	79.378	79.43	+.052
- 5	170	77.110	77.13	+.02
-10	160	72.574	72.62	+.046
-10	150	68.038	68.06	+.022
+50	200	90.718	90.76	+.042
+50	250	113.398	113.43	+.032
+50	300	136.077	136.10	+.023
+50	350	158.756	158.78	+.024
+ 5	355	161.024	161.04	+.016
+10	365	165.560	165.57	+.01
+10	375	170.096	170.11	+.014
+50	425	192.775	192.80	+.025
+50	475	215.455	215.45	+.005
+50	525	238.135	238.13	-.005
- 5	520	235.867	235.85	-.017
-10	510	231.331	231.29	-.041
-10	500	226.795	226.76	-.035
+50	550	249.475	249.46	-.015
+50	600	272.154	272.11	-.044
+50	650	294.833	294.81	-.023
+ 5	655	297.101	297.09	+.011
+10	665	301.637	301.60	-.037
+10	675	306.173	306.16	-.013
+50	725	328.853	328.84	-.013
- 5	720	326.585	326.57	-.015
+50	770	349.264	349.24	-.024
- 2 @ 10 and - 5 @ 50	500	226.795	226.79	-.005
- 5 @ 50	250	113.398	113.41	+.012
- 5 @ 50	0	0	-.01	-.01

Allowable Difference ± 0.07 Kg (70 grams)

Scale No. 01270
Date Calib. 2/5/80
Time 10:00
Operator [signature]

* + = place weights on scale
  - = remove weights

** Zero scale before starting

POOR ORIGINAL

ATTACHMENT NO. 2
PAGE 1 OF

AGNS WORK REQUEST

CHARGE NO. 30091 PROJ NO: AWR NO: 14416

FACILITY			DEPARTMENT		CATEGORY		OTHER REQUIREMENTS		MAN HOURS ESTIMATED/ACTUAL
Separations	(1)	X	Production	X	Q	X	None	X	/
UF_6	(2)	☐	Analytical	☐	Non-Q	☐	RWP	☐	/
Plutonium	(3)	☐	Electrical	☐	R	☐	HWP	☐	/
Waste	(4)	☐	Instrument	☐	NRed	☐	CA	☐	/
Utilities	(5)	☐	Mechanical	☐			CTA	☐	/
			QC	☐					/

REQUESTOR: J.D. Logue 2/15/80 EXT. 463 DATE REQUIRED: 2-18-80
SUBJECT: EQUIP. NO. 012 LOCATION: FRSS
DESCRIPTION OF WORK:

Additional Instructions for CNSI-TP-02, Mockup Filter Ballast and Drying Test

1. Complete steps 8.1 thru 8.11 of CNSI-TP-02.
2. Make additional runs as follows:
   2.1 Set FI at 10 SCFM for blowdowns.
   2.2 Complete data sheet and blowdown as follows: 1&2 blowdown 10 mins; 3&4 blowdown 2 hrs; 5&6 blown 4 hrs; 7&8 blowdown 8 hrs; 9&10 blowdown 24 hrs.

DATA SHEET

	Run	Tare	Gross wt	Amt left	Blowdown	Drain	Start	Stop	
Pre-filter	1	34.23	314.82	38.34	313.41	1350ml	10:51	11:15	* at 5 SCFM
	2	34.23	312.77	39.63	312.21	290ml	11:45	12:03	
	3	34.23	308.78	35.37	308.23	80ml	12:37	14:45	
	4	31.26	313.83	39.93	313.33	70ml	08:15	10:36	
	5	31.26	313.98	38.89	313.75	30ml	11:15	15:50	
	6	31.21	311.09	49.30	309.99	22ml	16:45	21:00	
	7	31.21	312.40	43.49	311.79	9.5ml	08:00	16:15	
	8	31.31	~~311.08~~ 305.16	~~41.32~~ 35.30	~~310.45~~ 304.91	13ml	07:00	15:15	
	9	33.62	311.08	42.32	310.45	0.0	1600	15:25	
	10	33.90	318.87	41.22	317.77	0.0	1600	0800	+16 hrs
Final Filter	1	33.62	295.68	51.00	293.05	2156ml	10:30	10:48	
2hrs *	2	33.62	296.15	49.72	295.46	180ml	11:27	13:35	* 2hrs
10min	3	33.90	317.77	71.5	314.50	1920ml	0330	0902	10min
	4	33.90	313.34	68.65	310.92	175ml	0845	1100	
	5	33.90	314.95	73.59	313.70	153ml	1130	1340	
	6	33.90	310.92	65.63	309.70	140ml	1118	1530	
	7	34.26	313.58	93.6	312.86	22ml	1545	0908	
	8	34.38	309.72	74.03	309.37	31	0900	1615	
	9	33.90	316.70	71.95	314.34	0	1440	1510	
	10	34.29	311.86	78.[illegible]	309.66	0	0900	0930	

APPROVAL TO COMMENCE WORK: Cognizant Supervisor [illegible] 2/18/80 Time: 0830 Date 2/18/80

APPROVALS: Cog. Engineer JDL 2/15/80

PLANNER: [illegible] 2-18-80 QA NRG

SLC: [illegible]

OSC: NRG

AWR Complete: AWR Follow-up Req. ☐ Yes ☒ No (If yes, complete back of form)

Tech. [illegible] Time 1100 Date 2/29/80

Cog. Supervisor JDL

P.D. Supervisor:

Form OO-15 Distribution: White—AWR File Rec. Mgt., Yellow—performing dept., pink—performing dept./planner/originator, green—originator

Page 1 of 1

POOR ORIGINAL

DATA TABLES

TABLE 1

WEIGHTS, CAPACITIES, AND DRY STORAGE TEST RESULTS

Filter Vessel	Weights		Weight Required for Ballast	Dry Storage		
	Empty	Capacity		Time	Press	Liquid
Pre	522 lbs.	274.3 ℓ	381[1]	48	15	None
Final	678 lbs.	245.1 ℓ	381[1]	48	15	None

---

(1) Maximum Weight - Limited space and weight configuration prevented adding additional weight.

TABLE 2

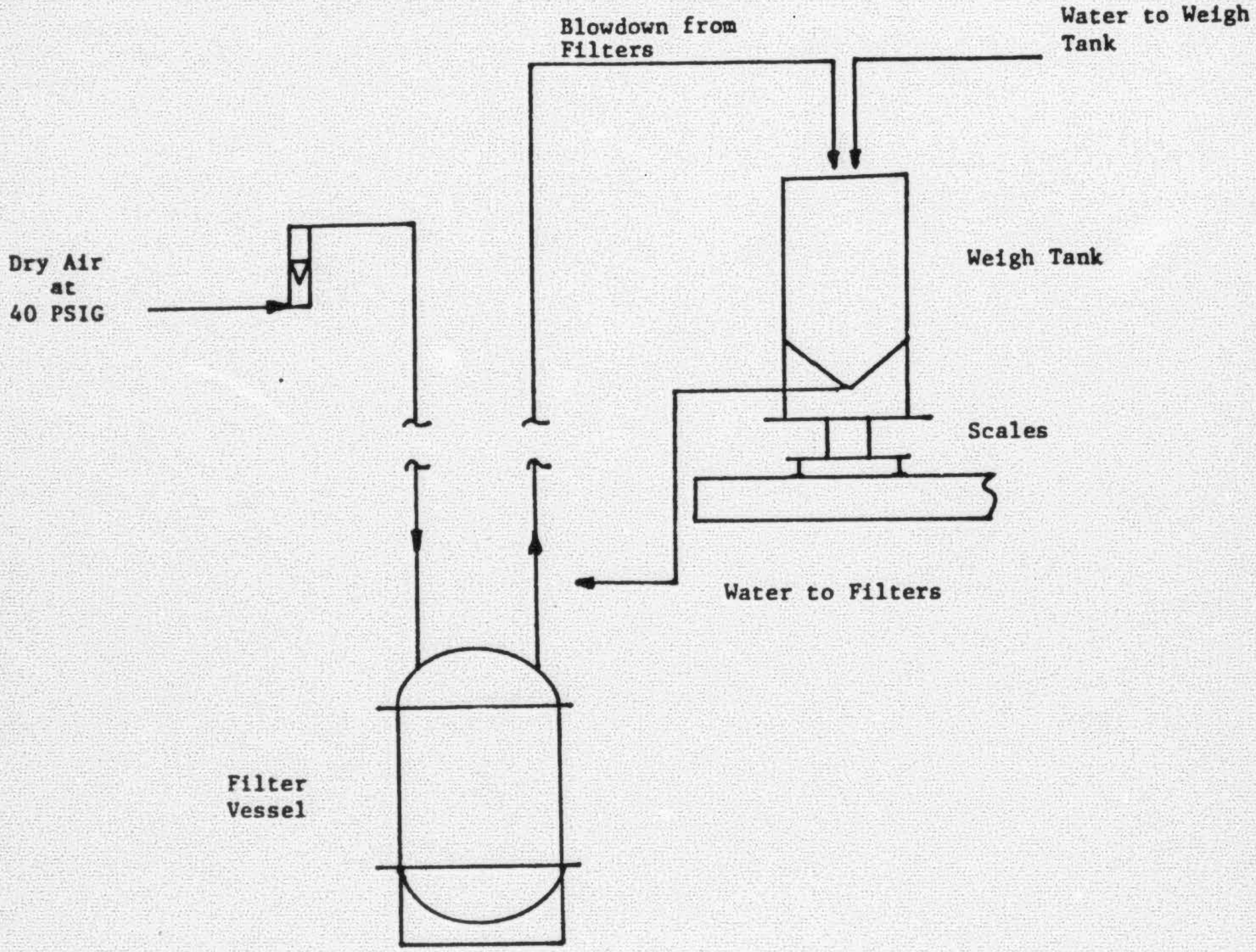
BLOWDOWN TIMES

Blowdown Times	Volume Remaining			
	Prefilter		Final Filter	
	Run 1	Run 2	Run 1	Run 2
10 minutes	1350 ml[1]	290 ml	2156 ml	1920 ml
2 hours	80 ml	70 ml	180 ml	175 ml
4 hours	30 ml	22 ml	153 ml	140 ml
8 hours	9.5 ml	13 ml	22 ml	31 ml
16 hours[2]	0	0		
24 hours			0	0

---

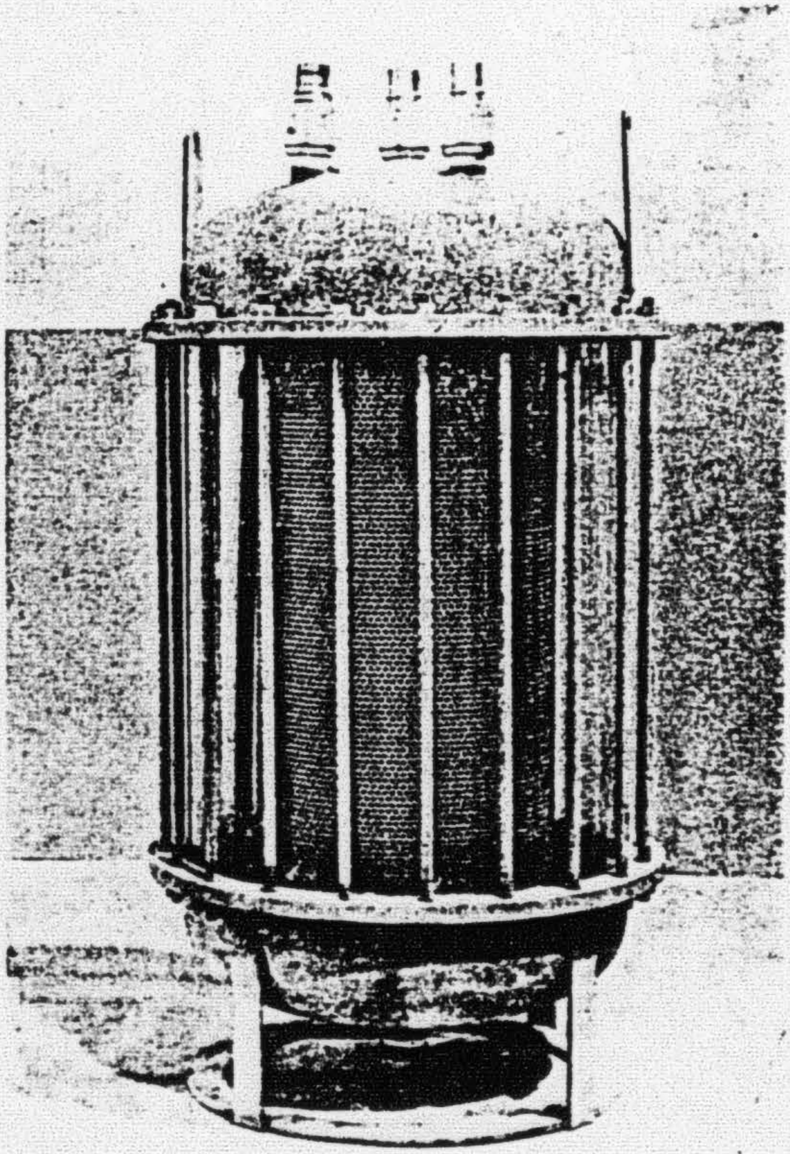
(1) Blowdown was done at 5 scfm.

(2) Visual inspection of the final filters during the 24-hour blowdown indicated wetness of the filter after 16 hours.



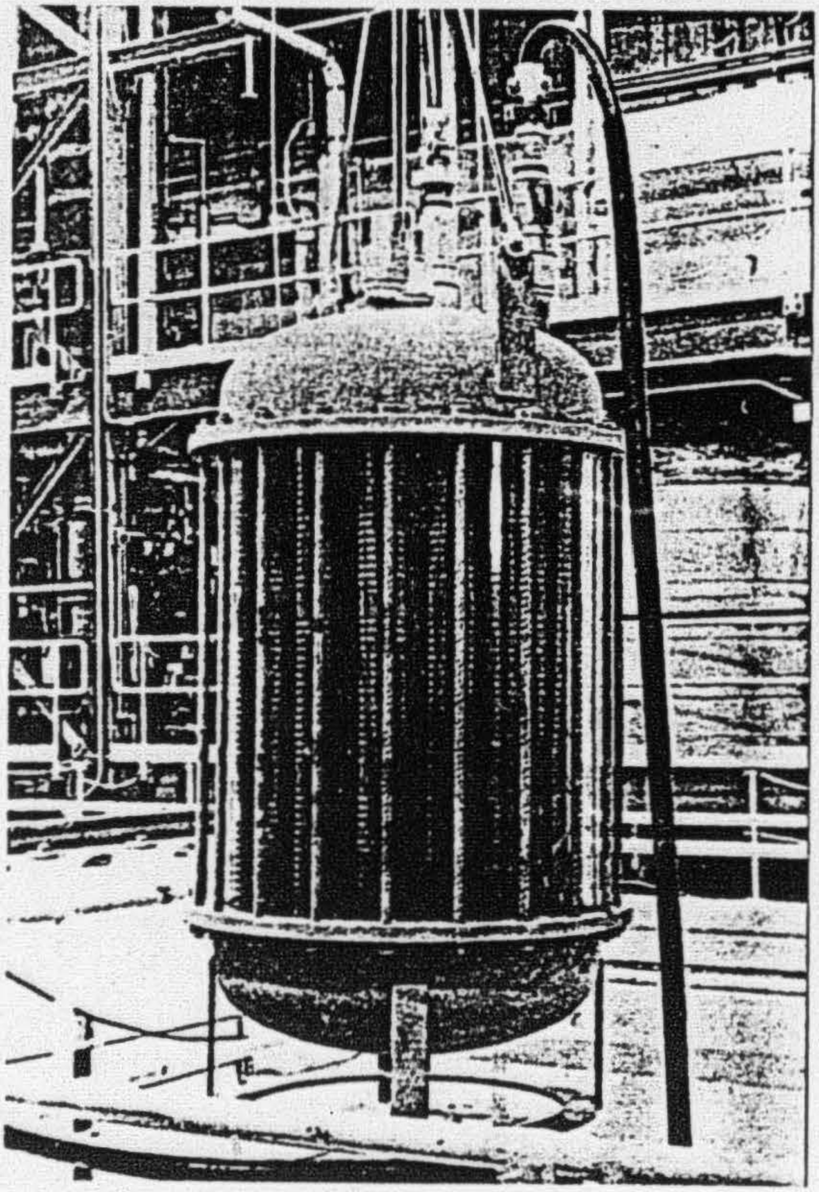
BLOWDOWN SETUP FOR CNSI-TMI FILTERS

FIGURE 1



CNSI TMI PREFILTER TEST UNIT

FIGURE 2



CNSI TMI FINAL FILTER TEST UNIT

FIGURE 3

SPECIFIC COMMENTS

QUESTION 22

The SDS feed tanks can be considered as one feed tank with a capacity of 60,000 gallons since there are no valves between these feed tanks and two monitor tanks, which receive processed waste water have a capacity of 19,000 gallons each.

Describe the criteria for determining the batch volume without risking radioactivity contamination of already processed water in the monitor tanks. The in-line monitors may not provide the sensitivity required to detect desired radioactivity breakthrough due to the radioactivity plateout and background radiation buildup.

RESPONSE

It is anticipated that 200 bed volumes (approximately 12,000 gallons) will be processed prior to zeolite bed removal and replacement. Following SDS shutdown, the nearly filled monitoring tank will be isolated and the empty tank valved in prior to startup.

The beta monitors will be monitoring the effluents of the first, second, and third zeolite beds. Additional monitors are provided for the cation influent and effluent as well as the in-line monitor before the polishing unit. These detectors coupled with the routine sampling and analyses program will provide early detection of increasing levels at various points in the process stream. All beta monitors view a thin-walled section of piping that may be removed and replaced, should plateout restrict sensitivity.

SPECIFIC COMMENTS

QUESTION 23

Provide estimated contact dose on spent filter, zeolite, and cation vessel cask.

RESPONSE

Activity loadings of the filter units has been estimated based on the expected activity of filterable solids in the 125 micron and 10 micron range and includes the amount of residual liquid remaining after dewatering. The coarse filter has been estimated to have a contact dose reading of approximately 3 R/hr. and the final filter to have a contact dose reading of approximately 400 mr/hr.

The contact dose rate on the zeolite bed has been estimated as 30,000 R/hr., based on a loading of ∿ 9400 Ci of Cesium. The zeolite bed will only be transferred with the shield cask in place. The contact dose rate on the shield cask is less than 1 R/hr. with the above curie loading.

Column testing has indicated very little ion exchange of radionuclides on the cation bed, and, therefore, insignificant contact dose rates will be present.

SPECIFIC COMMENTS

QUESTION 24

You have provided an in-line monitor in a common recirculation line from all vessel containment boxes to the leakage containment ion exchangers. Describe leakage detection method to identify the source of leakage from each vessel containment box.

Describe also how you would distinguish accidental leakage from the leakages resulting from routine connections and disconnections of vessels.

RESPONSE

As indicated on P & ID's 527D-A-5001 and 5002, each leakage containment is provided with an isolation valve on the exit line. In the event of a gross leak during system operation, the source may be isolated by selective valving and observation of the in-line monitor.

This monitor was provided to detect significant leaks during system operation. It is provided with an automatic time delay isolation of the SDS feed. In view of the fact that submerged components will be flushed prior to disconnect and the large leakage containment dilution flow, it is anticipated that this monitor will be unaffected by vessel uncoupling.

SPECIFIC COMMENTS

QUESTION 25

Describe leakage detection method provided within Hi-Rad Filter Glove Box, Hi-Rad Feed Sample Box, Intermediate Level Sampling Glove Box, and Beta Monitor Manifold. Describe (1) type of analysis to be performed on Hi-Rad samples, and (2) capability of such analyses on the TMI site.

RESPONSE

The Hi-Rad Filter, Hi-Rad Feed, Intermediate Level Sampling boxes, as well as the Beta Manifold, are contained shielded enclosures that are connected to drains and the SDS offgas system. In view of the above, no leak detection system is provided. When the SDS is operating, the area will be under constant surveillance and these enclosures may be visually inspected for leakage. In addition, a leak may also be indicated by a rise in the SDS offgas monitor. Routine radiological surveillance, in addition to the installed instrumentation, will provide early detection of minor leakage.

The types of analyses anticipated to be performed on Hi-Rad samples are as follows:

1) Suspended solids
2) Gross Beta
3) Gamma spectral analyses
4) Strontium 89 and 90
5) pH
6) Conductivity
7) Specific ions

Sufficient chemical equipment will be installed to perform these studies within the shielded sample boxes. High level samples will also be diluted within the boxes prior to movement to chemical work areas for preparation of samples for counting. Counting equipment has been purchased for dedicated use in support of SDS operation. These include gas flow counters, end window G-M's, and a computerized Ge(Li) System.

SPECIFIC COMMENTS

OUESTION 26

You state that the capability to solidify the spent resin vessels generated by the operation of the SDS has been developed. Describe the capability and process involved in detail.

RESPONSE

The attached drawing number 527-D-S-1000, indicates the preliminary concepts for a 10ft^3 vessel solidification facility. Current plans are to install this facility in the loading dock area of the Fuel Handling Building truck bay. The cubicle will be provided with a filtered exhaust system that will be ducted to existing station ventilation. Additional features include closed circuit TV cameras, radiation monitors, and a motor driven cask cart.

A dewatered vessel will be loaded underwater in the CNSI 1-13C cask and allowed to drain prior to movement to the solidification cubicle. When inside, the cask lid will be removed and the remote manipulator will be used to connect water, cement, and vent lines. The proper amount of water and dry cement will be metered to the vessel and internal agitator blades will be rotated to assure proper mixing. Following remote disconnect, the cask lid will be installed and the cask will be moved outside the cubicle for final preparation.

Pending further guidance and criteria development for the ultimate disposal of spent ion exchange media by the NRC, we have discontinued further work on the engineering design details and development of the SDS solidification facility.

SPECIFIC COMMENTS

QUESTION 27

Provide an estimate of dissolved KR-85 concentration and include it as part of the gaseous source term.

RESPONSE

Assuming the containment Kr-85 airborne concentration is 1 μCi/ml, the resulting water concentration is approximately 5×10^{-2} μCi/ml, the SDS process rate is 10 gpm, and that all Kr-85 is released to the offgas system during processing, the resulting Kr-85 release rate would be 1.82×10^{3} μCi/minute. Including this contributor to the gaseous source term does not significantly alter the conclusions presented in Chapter 6 of the SDS TER. The maximally exposed individual would receive an additional exposure to the lungs of 8.54×10^{-4} mrem per year and 9.96×10^{-2} mrem per year to the skin, the critical organs for this isotope. Additional annual exposures to other organs, listed below, are insignificant.

Bone:	6.5×10^{-4} mrem	Thyroid:	4.46×10^{-4} mrem
Total Body:	5.6×10^{-4} mrem	GI Tract:	4.42×10^{-4} mrem

SPECIFIC COMMENTS

QUESTION 28

Wouldn't the compartment cleanup and monitoring system for the pool water indicate a leakage contamination problem long before area radiation monitors in the SDS area would detect such leakage? What level are these area monitor alarms set? Provide type, number, location, and setpoint for all area radiation monitors.

RESPONSE

It is agreed that the leakage containment system will observe in-leakage from the spent fuel pool to the containment boxes. However, in the event of a leak on the effluent line, the time response of this radiation monitor relative to other area monitors is sensitive to the location of the assumed leak.

The area monitor associated with the SDS which would detect an increase in spent fuel activity are listed below:

INSTRUMENT NUMBER	TITLE	VENDOR	RANGE
CN-RE-1X03	SDS Manifold General Area Radiation Element	Eberline	1-10,000 mR/hr
CN-RE-VA06	SDS Offgas Header Influent Radiation Element	Eberline	.01-100 mR/hr
CN-RE-LC05	SDS Leakage Containment System Influent Radiation Element	LND-INC	$10^1 - 10^6$ CPS

The current plan is to set the alarm setpoints slightly above normal operational levels.

SPECIFIC COMMENTS

QUESTION 29

State if the reactor building water pump and SDS feed pump are provided with automatic trip upon detection of a present high radiation level by in-line radiation monitor (RE 527-13), fuel pool area monitors, and/or radiation monitors in the filter containment boxes.

RESPONSE

A high radiation indication at the in-line radiation monitor will cause valve V527-21 to close. This valve isolates the feed to both ion exchanger trains. When this occurs, the SDS feed pump 527-G-01 goes into a bypass mode. The reactor building water pump (WGP-1) is not tied into the in-line radiation monitor RE-527-13.

Fuel pool area monitors and filter containment box radiation monitors are alarmed only.