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4410-84-L-0094
Document ID 0015A

July 19, 1984

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

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Makeup and Purification Cesium Elution
Safety Evaluation Report

Attached for your review and approval is a copy of the Safety Evaluation Report (SER) for the Makeup and Purification System Resin Cesium Elution.

This SER incorporates, to an appropriate level, the issues discussed in Mr. L. H. Barrett's letter to GPU Nuclear dated May 25, 1984.

In accordance with 10 CFR Part 170, enclosed is a check for an application fee of $150.00.

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

Sincerely,

B. K. Kanga
Director, TMI-2

w/ check $150
#: 00119781

D009
1/1



BKK/SDC/jep

Attachment: Check No. 00119781

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

GPU Nuclear

SAFETY ANALYSIS

SA # 4340-3211-84-0054

Rev. # 0

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TITLE

SAFETY EVALUATION REPORT
FOR

Rinse and Elution of Makeup and Purification Demineralizer Resins

# SAFETY EVALUATION REPORT
# FOR
# RINSE AND ELUTION OF MU & P DEMINERALIZER RESINS

SAFETY EVALUATION REPORT
FOR
RINSE AND ELUTION OF MU & P DEMINERALIZER RESINS



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## 1.0 PURPOSE, SCOPE AND ORGANIZATION

### 1.1 Purpose

The purpose of the SER is to demonstrate that the evolutions comprising the rinse and elution of the TMI-2 Makeup and Purification System demineralizer resins can be accomplished without presenting undue risk to the health and safety of the public.

The objective of the rinse and elution project is to remove a majority of the activity from the resins while they are in the demineralizers to the extent that standard resin sluice procedures can be carried out.

### 1.2 Scope

The scope of this evaluation includes only the first phase of an essentially three phase process for disposition of the Makeup and Purification resins. This first phase includes the rinse and elution of the demineralizer resins; the sluicing, removal, solidification or other packaging and disposal of these resins are included in the latter two phases and will be addressed in separate SER's.

### 1.3 Organization

Section 2.0 contains a description of the rinse and elution process as well as a description of the hardware utilized to perform this process.

Section 3.0 describes the radiological considerations, including the expected radiation dose rates and engineering features designed to keep exposures as low as reasonably achievable (ALARA).

Section 4.0 addresses safety concerns and design features of the process. Section 5.0 presents the summary and conclusions of the safety evaluation. Section 6.0 provides the references.

## 2.0 DESCRIPTION OF THE RINSE AND ELUTION PROCESS

### 2.1 Background

The demineralizers of the Makeup and Purification System are located on elevation 305'-0" in the TMI-2 Auxiliary Building. The "A" and "B" demineralizers are essentially identical in design (See Table 1) and are located in separate cubicles. There is a floor drain in each cubicle which drains into the Aux. Building Sump. The demineralizer cubicles are shielded with four (4) feet of concrete in each wall, floor and ceiling. Penetrations between the Hayes Gas Analyzer Room and each demineralizer cubicle are located at elevation 321'-9" (2'-3" from the ceiling).

These demineralizers accumulated significant quantities of fission products and small quantities of fuel debris during and subsequent to the March 1979 accident.

Until recently, the physical condition of the demineralizers and their contents were unknown. The radiation and thermal exposure to the installed resins were expected to be sufficient to potentially cause severe degradation. However, the demineralizer resin samples have not shown significant damage.

Operations to decontaminate the demineralizers are complicated by the high radiation levels. As a result of the high radiation levels, the demineralizers and components within the cubicles are not available for personnel access. Additionally, if these resins were to be transferred by normal procedures, other plant areas and systems normally used for resin disposal would see radiation levels in excess of their design. Therefore, a method of reducing the activity on these resins is necessary.

## 2.2 System Hardware

The system consists of four equipment skids which will interface with existing plant systems. They are 1) the Chemical Mix Tank and Fill Pump Skid; 2) the Dilution Eductor and Transfer Pump Skid (one skid for each demineralizer); and 3) the Back Flushable Stainless Steel Filter Skid. (Refer to Figure 1.)

The following are the plant systems with which the skids interface:

1) Makeup and Purification
2) Liquid Waste Disposal
3) Electrical
4) Processed Water
5) Waste Gas Decay
6) Instrument Air

The specific interfaces between these systems and the elution skids are addressed in Section 4.1. Existing valves through which process flow is required are listed in Table 2.

## 2.3 Process Description

The cesium elution process utilizes a pump/eductor skid for each demineralizer which draws water from the respective demineralizer through the back flushable filter(s) and then feeds to the installed plant neutralizer tanks. The filters may be operated individually or in parallel. The "A" and "B" demineralizers will be processed separately. Additionally, the process is designed so that it is possible to stop processing from one unit and initiate processing from the other.

Currently the "A" and "B" Demineralizers contain water. Sample data of the water in the "B" demineralizer indicates that cesium levels may currently be as high as 2700 μCi/ml, whereas samples of the "A" Demineralizer indicate considerably lower cesium concentration. Additionally, the water level in each demineralizer is undetermined. Therefore, before a demineralizer is first processed, it will be necessary to establish the water level. In order to do this, a suction line from the pump/eductor skid will be used to access the demineralizer through the resin fill lines. The suction line will be cut to a predetermined length in order to provide a benchmark for water level in each demineralizer. This benchmark level should be controllable within ± four (4) inches or ± approximately 35 gallons.

Thus, by adding sufficient water to each demineralizer via the chemical addition skid to bring the water level above the end of the suction line, a known water level will be established. All water used for filling the demineralizers and for subsequent dilution will be borated water from the processed water system. A record of all water additions will be procedurally maintained. The capability to pump down to this benchmark will exist throughout the processing operation, thus providing the capability to always return to a known elevation.

Once the initial level is established in each demineralizer, two hundred (200) gallons of processed water will be added to the bottom of the demineralizer via the normal operational outlet header. This will dilute the concentration of cesium in the water currently contained in the demineralizer. Since this water will contain very low sodium concentrations, no elution of cesium from the resins should occur. This dilution is expected to yield a maximum cesium concentration of 1350 μCi/ml in the demineralizer. Processed water (i.e., dilution water) flow will then be initiated through the eductor at five (5) GPM. (This flow provides the motive force for initiating flow from the demineralizer by generating a suction on the influent of the eductor which communicates directly with the previously installed suction line in the demineralizer.) This results in a siphoning of the liquid from the demineralizer at approximately 0.25 GPM which provides a volumetric dilution of the demineralizer water by approximately 20 to 1 at the eductor effluent. With an eductor influent concentration of 1350 μCi/ml, this will result in water with concentrations of 70 μCi/ml being delivered from the pump/eductor skid to the backflushable filters located on the mezzanine. NOTE: Dilution rates up to 50 to 1 are possible if activity of the water is higher than expected.

Subsequent batches, up to approximately 200 gallons, of liquid pumped into the demineralizers from the chemical mix tank will be of processed water containing NaOH. This chemical addition will cause elution of cesium from the resins into the water introduced to the demineralizer.

The exact concentration will be determined within the limits set in the process instruction and data sheet. It is envisioned, however, that later batches will contain increasingly higher $Na^+$ concentrations.

When levels of activity are low enough to maintain effluent levels from the pump/eductor skid below 70 µCi/ml, direct pumping of liquid from the demineralizers will commence. The pump will deliver water from the demineralizers utilizing the same suction hose which was used with the eductor, at rates up to 3 gallons per minute. Processed water may also be used with the pump. Dilution will be adjusted to insure that activity levels in the piping/filtration system are maintained below 70 µCi/ml.

During initial operation, the system will utilize one (1) filter with the other filter on standby. The back flushable filters are designed to extract fuel fines and other solid debris greater than 20 µm at 5.3 GPM. Differential pressure across the filters as well as filter radiation levels will be monitored.

The back flushable filters have been designed so that as they become loaded with material, they can be backflushed to either demineralizer using 90 psig nitrogen. If the nitrogen backflush should fail to adequately clean the operating filter, the standby filter would be brought into service. Additionally, processed water is supplied to the backflushable filters for rinsing the piping between the filters and the demineralizers. This will flush any particulate which was removed from the filter by the nitrogen backflush.

From the filters, the effluent flows through a water meter assembly which has remote readout at the operating console. This water meter assembly also has provisions for the introduction of additional processed water for flushing and/or further dilution. The effluent of the water meter assembly is hard piped through an existing penetration into the mezzanine area of the valve alley corridor. From there, temporarily shielded hose delivers the effluent by the most direct path to the neutralizer tanks (WDL-T-8A and 8B) located at elevation 280'-6" in the Fuel Handling Building. The neutralizer tanks have a capacity of 9,640 gallons each. These tanks will act as holding tanks for further batch processing of the demineralizer effluent via the Submerged Demineralizer System (SDS) and/or EPICOR II.

The proof of principal for this process was done by Oak Ridge National Laboratories (ORNL), utilizing previously acquired resin samples, to determine elution rates and to identify chemical additives and concentrations necessary to achieve the desired elution rates.

The decision to stop processing the "A" or "B" demineralizer will be based on available data from the ongoing monitoring of the eductor/pump effluent. If activity levels are confirmed to be

sufficiently low to allow the resins to be transferred utilizing the existing in-plant resin transfer system, the process will be terminated. If, however, a point is reached where efficient removal of cesium ceases to occur prior to obtaining the stated goal, the processing will be stopped and the purification demineralizer plans will then be reevaluated. During shutdown or upon completion of the elution process, procedures will then be implemented as appropriate for monitoring and maintaining the resin environment and system integrity.

## 3.0 RADIOLOGICAL CONSIDERATIONS

### 3.1 ALARA Considerations

All personnel performing work in accordance with this evaluation shall utilize every means available to maintain their radiation exposure as low as reasonably achievable (ALARA). Radiation Control personnel will monitor work areas as required and provide dose rate information to aid individuals in performance of tasks in so far as radiological work practices are concerned.

Extensive planning of tasks to be conducted in a radiation field, and training of personnel may be used to reduce the time needed to complete a task. The higher radiation areas are identified to personnel and shielded where practicable. The work is structured to avoid these areas to the extent possible. This includes the location of the Control Panel behind an existing shield wall. In addition, the Chemical Addition Skid is located in the valve alley corridor to minimize exposure during the chemical mixing process.

### 3.2 Existing Contamination in Working Access Areas

Worker accessibility for processing of the purification demineralizers will be limited to the Hayes Gas Analyzer Room located immediately north of the demineralizer cubicles. The Hayes Gas Analyzer Room has been decontaminated to the extent necessary such that respirators will not be required for processing personnel.

### 3.3 Dilution

Curie levels for initial rinse of the demineralizers are anticipated to be as high as 1350 µCi/ml based on sample results from ORNL. Since the levels required for reasonable management through normal waste processing (i.e., to the Neutralizer Tanks and the Submerged Demineralizer System) is 70 µCi/ml, the demineralizer rinse effluent will pass through the eductor or pump and will be diluted. The first batch will be diluted by 20:1 and subsequent batches will be adjusted to maintain levels not to exceed 70 µCi/ml.

3.4 Radiation Monitoring

Portable radiation monitors will be used in detecting radiation levels in the demineralizer cubicles, in the filtering process equipment, and general working areas. Radiological information and relevant data from within the demineralizer cubicles will be obtained remotely.

3.5 Shielding

Process flow piping into the Hayes Gas Analyzer Room will be shielded to maintain the work station levels at or below 2.5 mR/hr. The lines from the Chemical Mix Tank into the demineralizer will not be shielded since only processed water will be in those lines. Inadvertent backflow could only occur with failure of two check valves simultaneously with other valve leak/failures.

The eductor/transfer pump skid will be shielded to maintain work station radiation levels at or below 2.5mR/hr. Process flow valves have reach rods which penetrate the shielding.

The eductor/transfer pump skids are supported by structural support members which are anchored to the demineralizer cubicle walls, and floor. These structural supports also hold shielding for interconnecting piping.

The back flushable filters utilize a steel shielding design.

As previously indicated, the system's design provides for backflushing the filters with nitrogen and rinsing with processed water in order to flush collected activity back to the demineralizer vessels.

Effluent from the filters is directed through a shielded water meter assembly then to shielded piping which leads to the valve alley corridor. From there the effluent will be delivered to the neutralizer tanks using rubber hose. The hose will be prefabricated in 50 foot sections with factory installed, locking quick disconnect fittings to facilitate disposal. The fittings will be wrapped in plastic using existing plant contamination control practices. This hose will be shielded to maintain acceptable radiation levels.

The final destination of the process flow is the Neutralizer Tanks (WDL-T-8A, -8B). The tanks are shielded by concrete walls and are access controlled by Radiological Control procedures. Introduction of 70 μ Ci/ml water in the Neutralizer Tanks will not exceed the design criteria of the shielding walls of that cubicle.

## 4.0 SAFETY

To accomplish the rinse and elution task effectively, process pathways have been chosen which minimize impact on other plant operations. This section describes some of the sensitive parameters and postulated accident scenarios which have been accounted for in the design.

### 4.1 Interface to Existing Piping

To incorporate the hardware skids into the elution process, there are five (5) tie-in points at which new piping will be temporarily connected to existing plant systems.

The tie-in point on the front end of the process, which introduces rinse water into the bottom of the demineralizers, is at the outboard flange of Valve MU-V114. Here, a 1" hose will be connected to an existing 2" flange at MU-V114.

The two process flow downstream tie-in points will be made at the existing resin fill line flanges. These tie-in points will be flanged connections to enable respective eductor bypass during backflush of the filters.

Two flanged connections are required to connect the process line to the WDL piping at the Neutralizer Tanks upstream of WDL-V373A, -V373B.

Each of these flanged connections will be of standard 150# ASA, Type 304L stainless steel. They will be sealed and bolted per ANSI B31.1, paragraph 104.5. (Reference 2)

The entire system of new process hardware and existing vessels/piping will be isolated from the rest of the plant systems by at least one barrier. Testing to the extent practical prior to processing will determine the integrity of these barriers or the need to provide additional isolation. Testing requirements are discussed in Section 4.2.

For accident scenarios that could credibly dilute the RCS below 3500 ppmB, double barrier isolation will be provided from the RCS during both processing and static conditions in the RCS.

### 4.2 Radioactive Release

All process flow piping from the pump/educator skid up to and including the hose will be hydrostatically tested per Regulatory Guide 1.143 and ANSI B31.1 in order to guard against line breaks, valve, pump and flange leaks. Operating pressure of the system is not to exceed 150 psig (during backflush). Therefore, the hydrostatic test will be performed at 225 psig for 30 minutes. Backflush of the stainless steel filters will not increase the pressure in the demineralizer to more than 10 psig. (Demineralizer

pressure will be maintained between 2 and 8 psig by the Waste Gas Decay System during normal operation of the process system.) Additionally, the new process hardware will be tested to the extent practical with processed water prior to elution to verify system operability and integrity. These tests are expected to identify system weaknesses so that they may be corrected in order to preclude a radioactive release during operation.

To reduce the potential for a large leak of 2700 µCi/ml water, the eductor has been placed at 318'-0" (thirteen feet above the floor). The top of the demineralizers are at 313'-4". Assuming a potential 8 psi overpressure in the demineralizer, a leak upstream of the eductor cannot result in siphoning more than 60 gallons of water out of the demineralizer and onto the floor.

An existing radiation monitor, HP-R206, is equipped with alarms which will sound in the event of an increase in general area radiation levels. In addition, Rad Con surveillance will be ongoing as required.

4.3 Dilution Control

Sufficient instrumentation is provided to determine and control demineralizer and dilution water flowrates such that effluent concentrations are at or below 70 µCi/ml. The instrumentation consists of a radiation monitor for the eductor effluent line and flowmeters for all dilution water supply lines as well as the feed and suction lines in the demineralizers.

Dilution of the demineralizer water will primarily be accomplished at the eductor or pump, however, the capability exists for additional dilution downstream from the filters. The eductor can be throttled to provide dilution ratios of 20:1 up to 50:1. Dilution ratios of less than 20:1 can be obtained using the transfer pump.

Flowrate and total integrated flow will be closely monitored so that fluid level inside the demineralizers can be calculated at all times (1 foot of water in the demineralizer = 100 gallons).

4.4 Hydrogen Generation

Hydrogen gas has been generated in the demineralizers as a result of radiolytic breakup of water in a radiation field. This generation rate has been measured in the "B" demineralizer at 1.4 ℓ/day, and has been calculated to be no more than 0.25 ℓ/day for the "A" demineralizer. This rate of hydrogen generation will not inhibit any phase of the operations relating to this rinse and elution process. Hydrogen generated inside the tank presents no detonation problem because the small generation rate coupled with removal of gas via the Waste Gas Decay System will not allow buildup of hydrogen to the unsafe level required for detonation.

Any hydrogen released from the tank into the cubicle will be removed by the existing ventilation system.

4.5 Criticality

Criticality is not a safety concern. The Hanford Energy Development Laboratory (HEDL) has shown (Reference 1) that there is a total of less than 4 Kg of fuel in the demineralizers. The most reactive (spherical) configuration of TMI-2 type fuel ( 3% $U^{235}$ enriched) in an optimal moderator/reflector array requires at least 93 Kg of fuel for criticality to be possible.

4.6 Unmonitored Liquid Release in the Auxiliary and Fuel Handling Buildings

Properly placed and calibrated general area radiation detectors will monitor, indicate and respond to any inadvertent releases due to valve misalignment, pipe or hose rupture, or demineralizer overflow. However, unmonitored release scenarios were contemplated without accounting for initial detection safeguards. The scenarios and their consequences are described below.

A) Demineralizer Overflow

When liquid addition is attempted for the elution of the resins, a miscalculation and subsequent overaddition has been postulated. Overflow from demineralizer chemical addition could result in a liquid release whose activity might range as high as 1,350 µCi/ml. This overflow could potentially follow any of three pathways from each demineralizer.

(1) The liquid could simply fill the line up to and through the eductor given that the inline valves are open. The consequences of this could be safely mitigated, i.e., no actual release would occur in the building, but higher than anticipated dose rates (still less than 100 mR/hr) would be seen in the hose running to the Neutralizer Tanks. Radiation monitors on the hose would detect this so that process water could be flushed through the line to dilute the high activity water in the Neutralizer Tanks. Additionally, building traffic control will be utilized during times of transfer to the Neutralizer Tanks in order to preclude inadvertent personnel exposure in the event of the transfer of high activity water.

(2) The liquid could fill the off gas line to the Waste Gas Decay System. If the liquid were introduced to the Waste Gas header, moisture would be diverted by an inline drain and the moisture separators on the Waste Gas Compressors to the Auxiliary Building Sump. It is unlikely that the introduction of small quantities of water to the Auxiliary Building Sump would be detected but the water in this sump is normally

processed via the SDS System and stored on-site. No inadvertent release of this water would occur. Personnel exposure would be minimized since the sump is in a locked, High Rad Area.

(3) The third overflow pathway which was considered is backflow through the demineralizers' normal operation inlet headers. Liquid passing through this line could be released from piping in the Make-Up Filter Cubicle. These Make-Up Filters (MU-F-5A, -5B) have been removed and their housings left open, so a release has been postulated at this location. Liquid introduced to these filters would overflow the housing and spill onto the floor. There is a floor drain next to each filter housing which drains to the Auxiliary Building Sump. It is unlikely that the introduction of small quantities of water to the Auxiliary Building Sump would be detected but the water in this sump is normally processed via the SDS System and stored on-site. No inadvertent release of this water would occur. Personnel exposure would be minimized since the sump is in a locked, High Rad Area.

B) Piping and Valve Leaks

(1) Piping and/or valve leaks are possible in any area in which process piping will run. The areas on the 305' elevation that contain elution equipment/piping are the demineralizer cubicles, Hayes Gas Analyzer Room and the Make-Up Valve Alley. The associated areas on the 280'-6" elevation are the Neutralizer Tank Cubicle and the open corridor areas in the northern half of the Fuel Handling Building. The floor drains in these areas lead to the Auxiliary Building Sump. Since this type of upset condition has the potential to divert larger quantities of process water to the Auxiliary Building Sump than do the overflow conditions discussed in Section (A), it is more likely this type of leakage will be detected in the Sump. However, as water from the Auxiliary Building Sump is normally processed via SDS and stored on-site, no release to the environment would occur.

(2) A leak has also been postulated at demineralizer isolation valves MU-V107A and B, introducing contaminated water to MU-V8. MU-V8, in the RCS letdown pathway, is normally left in the "bleed" position, as a result contaminated water would be directed to the "C" bleed tank (normally used for RCS letdown). Liquid in the "C" bleed tank is processed through SDS, so that no release to the environment would occur.

4.7 10 CFR 50.59 Evaluation

Changes, Tests and Experiments, 10 CFR 50, paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test or experiment is determined not to be an unreviewed safety question and does not involve modification of the plant technical specifications.

A proposed change involves an unreviewed safety question if:

a) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b) the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) the margin of safety, as defined in the basis for any technical specification, is reduced.

The following paragraphs are the results of the 50.59 review that was performed for the resin rinse and elution activities.

The planned activities will not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the TMI-2 FSAR. The attachment to the SER provides an analysis of the maximum hypothetical accident (MHA) for the Makeup and Purification Demineralizers. This accident consists of a release of the entire contents of either demineralizer to the demineralizer cubicle and assesses the potential on-site and/or off-site radiological consequences of this accident. As this accident is similar to that described in Section 15.1.22 of the TMI-2 FSAR, "Small Spills or Leaks of Radioactive Material Outside Containment", a new type of accident has not been created. In addition, the consequences of this accident are within the bounds of those established in Section 15.1 22 of the TMI-2 FSAR. That section uses the Loss-of-Coolant Accident of Section 15.1.14 as the bounding case for environmental consequences, therefore, the probability of occurrence or consequence of this type of accident is not increased.

The TMI-2 Technical Specifications were also reviewed and it was determined that the margins of safety as defined in the basis for any Technical Specification will not be reduced.

Therefore, based on the evaluation presented above, it is concluded that the demineralizer rinse and elution evolution described in this SER does not involve an Unreviewed Safety Question as defined in Section 50.54 of 10 CFR Part 50.

## 5.0 CONCLUSIONS

Based on the radiological and safety evaluations contained in this report, the following have been demonstrated:

- Occupational exposure will be maintained ALARA.
- Demineralizer rinse and elution water will be diluted so that existing waste processing systems can be utilized.

- The process will be isolated from unrequired systems so that unnecessary contamination of other systems will be avoided.
- The consequences of potential radioactive releases are bounded by previous analyses.
- Radiolytic gas generation will not present any hazard.
- There is an insufficient amount of fuel and fissionable transuranics present to cause a criticality hazard.

## 6.0 REFERENCES

Reference 1. HEDL-7335, "Resin and Debris Removal System Conceptual Design", March 1983.

Reference 2. ANSI-B31.1 - 1973, "American National Standard Code for Pressure Piping".

Reference 3. GPUN Drawing No. 2E-3233-21-0002, "P&ID of MakeUp and Purification Cesium Elution System", 04/05/84, Rev. 0.

Reference 4. Letter from B. K. Kanga (GPUN) to B. J. Snyder (NRC), "Submerged Demineralized System Technical Evaluation Report." 4410-83-L-0122, July 6, 1983.

Reference 5. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants", Revision 1, October 1979.

TABLE 1

DEMINERALIZER SPECIFICATIONS

| | |
|---|---|
| Height = 83 inches - | filled with 50 $ft^3$ 2:1 cation-anion mixed bed organic resin. Tank is elevated 15" off the floor. |
| Diameter = 48 inches | |
| Normal Operation Inlet Header - | twelve distribution pipes 3/4" x 18" each with one end plugged. Each pipe has 21 3/16" holes. |
| Normal Operation Outlet Header - | Similar to inlet header pipes except the outlet header has 3/4" x 20" pipes with 110 mesh stainless steel screen. |
| Resin Addition Lines: | 3" normal operation inlets - via MU-V111A,B |
| | 3" normal operation outlets - via MU-V108A, B |
| Sampling Lines: | Normal operation inlet - via MU-V104<br>Normal operation outlet - via MU-V106 |

TABLE 2

VALVES REQUIRED FOR PROCESS FLOW

| Valve | Type | Location |
|---|---|---|
| MU-V111A,B | Grinnell Diaphragm operated | Just above each respective demineralizer |
| MU-V292A,B | Air driven globe valves | MU&P Valve Alley |
| MU-V293A,B | Check valves | MU&P Valve Alley |
| MU-V114 | Velan globe valve | MU&P Valve Alley Corridor |
| MU-V331 | Check Valve | MU&P Valve Alley Corridor |
| WDL-V-369A,B | Crane check valves | Neut. Tank Room |
| WDL-V373A,B | Grinnell Diaphragm Valves | Neut. Tank Room Fourier |

# SCHEMATIC FOR RINSE AND ELUTION

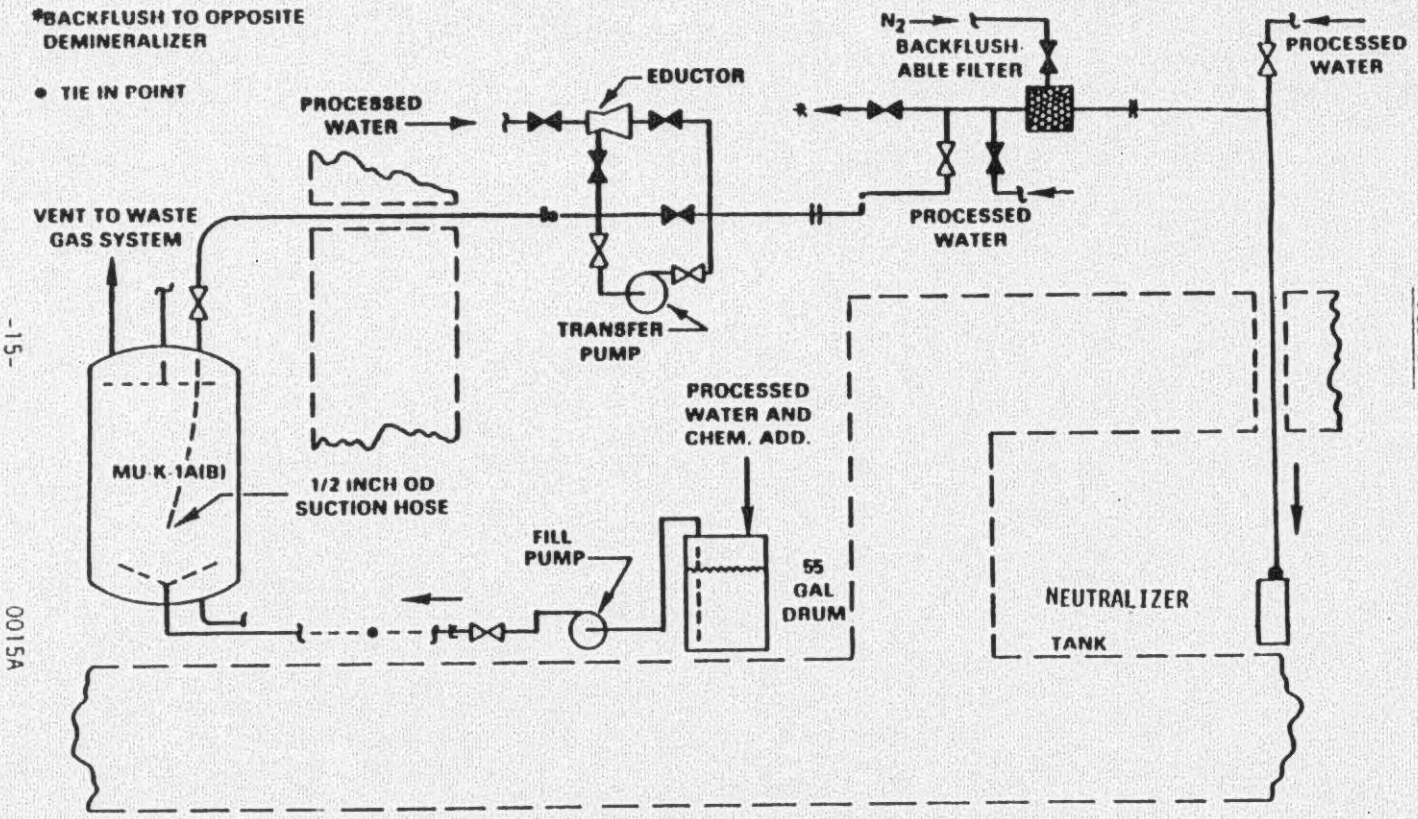


FIGURE 1

## MAXIMUM HYPOTHETICAL ACCIDENT

Preliminary - The purpose of this analysis is to determine the potential on-site and off-site radiological consequences due to an accident that would release the contents of the MU&P Demineralizer.

Maximum Hypothetical Accident (MHA) - Presently, the Makeup and Purification Demineralizers contain not only the greatest concentration of radioactive material but they contain the greatest total amount of radioactive material of any location at TMI-2 outside of the Unit-2 Reactor Building. Because these demineralizers were actively used at the time of the accident when finely divided fuel material was entrained in the RCS flowing through these demineralizers, the demineralizer resins contain fuel material and the associated transuranic alpha activity. It has been assumed, for this analysis, that the MHA occurs while the demineralizers contain their presently contained amounts of radioactivity.

The MHA is then considered to be the release of the contents of the demineralizer (22 ft.$^3$ of resin and 200 gallons of contaminated water) to the demineralizer cubicle. The demineralizer chosen for analysis is the "B" demineralizer which is believed, from a number of analyses, to contain the following curie content:

"B" Demineralizer Contents (Curies)
(as of 10/84)

| Isotope | Curie Content | Isotope | Curie Content |
|---|---|---|---|
| Cs-137 | 9020[a] | U-236 | 2.1E-5[m] 2.0E-5[c] |
| Cs-134 | 605[a] | U-237 | NM 2.1E-5[c] |
| Sr/Y-90 (each isotope) | 418[a] | U-238 | 1.5E-4[m] 1.5E-4[c] |
| Ru/Rh-106 " " | 5.1[b] | Np-237 | NM 6.8E-6[c] |
| Ce/Pr-144 " " | 5.0[b] | Np-239 | NM 9.6E-7[c] |
| Sb-125 | 29.5[b] | Pu-236 | NM 3.8E-7[c] |
| Te-125m | 7.1[b] | Pu-238 | < 8.6E-3[m] 2.8E-5[c] |
| I-129 | 6.44E-3[b] | Pu-239 | 5.66E-2[m] 5.13E-2[c] |
| Th-234 | NM 1.5E-4[c] | Pu-240 | 1.8E-2[m] 1.29E-2[c] |
| Pa-233 | NM 6.8E-6[c] | Pu-241 | 1.4E+0[m] 8.7E-1[c] |
| Pa-234m | NM 1.5E-4[c] | Pu-242 | < 1.9E-6[m] 1.2E-6[c] |
| U-234 | 6.7E-4[m] 7.9E-7[c] | Am-241 | NM 7.9E-3[c] |
| U-235 | 2.4E-5[m] 2.3E-5[c] | Am-243 | NM 9.6E-7[c] |

a-values calculated from analytical results of µCi/g and assuming a resin density of 0.75 g/cc and a total volume of 22 ft.$^3$
b-values believed to be in resins but masked by the cesium quantities
c-calculated based on ORIGEN values assuming 460 grams of fuel is contained in the demineralizer.
NM-not measured
m-measured for April 1983 sample in ppm & converted to curies based on 460 grams of fuel contained in the demineralizer.

It is interesting to note that for the April 1983 sample, the measured $\frac{\text{Total U (PPM)}}{\text{Total Pu (PPM)}}$ equalled 2.2E-3. The same ratio value, $\frac{\text{Total U (g)}}{\text{Total Pu (g)}}$, as calculated by the ORIGEN computer code was a very similar 1.96E-3.

Release of the contents of the demineralizer would amount to $1.38 \times 10^6$ cc of material. The floor drain inside the cubicle is assumed to be plugged by the resin material spilled from the tank. Result of the spill has been assumed to leave 50% of the resin inside the cubicle and 1 cm. of contaminated water on the floor. The remaining volume of resin, 11 $ft.^3$, and the remaining contaminated water is assumed to spill into the area adjacent to the cubicles which is essentially an area of 18 ft. by 42 ft. Water, other than a pool 1 cm. deep in this area, is assumed to be drained to the Auxiliary Building sump by the floor drain in the area outside the cubicle.

Releases of airborne activity to the Auxiliary Building during the spill would be negligible to the area outside of the cubicle. The wet resin and the dynamics of the spill will prevent all but very minor amounts of activity released to become airborne. Activity that does become airborne will be removed from the "B" demineralizer cubicle by the Auxiliary Building ventilation system.

After the spill, the water remaining on the floor of the cubicle and adjacent area is assumed to eventually drain into the sump. Cleanup of the residual activity is assumed to take place at some time after because of the high radiation fields that would exist in the area. During the intervening period of time the spilled activity associated with the degraded resin would again have to be wet down to prevent significant airborne activity from being generated during the cleanup operations.

Radiological Consequences of MHA (On-Site)

On-site radiological consequences of the MHA spill have been analyzed to determine exposure rates for two conditions:

a) Exposure rate at the opening of the enclosure adjacent to the cubicle due to the pool of contaminated water and resin on the floor, and
b) Exposure rate from the Auxiliary Building exhaust duct HEPA filters due to capture of the released airborne activity.

Pool Water Radiation Fields

Under the assumptions stated earlier, half of the resin volume and a pool of contaminated water 1 cm. deep remain within the demineralizer cubicle and the remaining material spills into the area outside the cubicle. This area is basically a 42.9' X 18.3' enclosure entered by way of a well shielded labyrinth. The area is served by two floor drains that more than likely would allow the spilled water to drain into the Auxiliary Building sump. For calculational purposes, however, this area of the floor is assumed to be left with a 1 cm. deep pool of water and resin. The area was modeled for the ISOSHLD-II computer code for the purpose of calculating the radiation field due to this pool. Labyrinth walls were excluded from the model such that it was considered to be a rectangular area 42' X 18' which contains 4990 curies of the primary gamma emitting isotopes, Cs-137, and 217 curies of Cs-134. Radiation field

values were calculated at three locations at 3 ft. above the pool: 1) over the center, 2) at the midpoint of the side, and 3) at the midpoint of the end. These radiation field values were calculated to be:

over the center - 297 R/hr
at the side midpoint - 193 R/hr
at the end midpoint - 163 R/hr

In the event all the water had drained away leaving only the resin, the corresponding radiation fields would be 79% of those presented above.

Auxiliary Building Exhaust Plenum HEPA Filter Radiation Fields

As mentioned in the SDS SER, a dropped SDS liner was assumed to release $10^{-4}$% of the contained activity to the Fuel Handling Building atmosphere. Because the spill considered for this analysis consists of a water-resin slurry, and because the spill is the result of a less violent event than a cask drop, only $10^{-6}$% of the activity is assumed to become airborne and entrained into the exhaust flow. Therefore, the resulting primary gamma emitting isotopic quantities of $9.02 \times 10^{-2}$ mCi of Cs-137 and $6.1 \times 10^{-3}$ mCi of Cs-134 are assumed to be loaded on the HEPA filter bank.

In December 1981, an analysis was performed on the Reactor Building Purge Exhaust duct HEPA filter bank. At that time this bank of HEPA's was determined to contain a total of 7.4 to 9.1 mCi of Cs-134 and 65 to 79 mCi of Cs-137. The resulting radiation readings were 10 mR/hr at the plenum side, 25 mR/hr at the plenum top, and an average of 64 mR/hr inside the plenum at the face of the filter bank. At the time this analysis was performed, the $\frac{\text{Cs-137}}{\text{Cs-134}}$ ratio equalled 8.8. Therefore, scaling the above analytical results to account for the activity released during the spill from the MU&P demineralizer, the $9.02 \times 10^{-2}$ m Ci of Cs-137 and $6.1 \times 10^{-3}$ m Ci of Cs-134 would add the following increases to whatever was present on the filters at the time of the spill:

Δ mr/hr at the plenum top = .03 mR/hr
Δ mr/hr at the plenum side = .01 mR/hr
Δ mr/hr at the filter bank face = .07 mR/hr

Therefore, these increases would not cause a significant increase in the exposure rate due to the HEPA filter bank.

Radiological Consequences of MHA (Off-Site)

For determining off-site radiological consequences the following assumptions are applicable to the release:

a) $10^{-6}$% of the demineralizer activity is released to the Auxiliary Building as airborne activity
b) the release occurs over a period of 15 seconds
c) a DF of 100 is assumed for the Auxiliary Building exhaust HEPA's.

Therefore, off-site activity release parameters are as given in the table below.

Table I

Off-Site Radioactivity Release
Parameters for MU&P System MHA

| Isotope | Station Vent Release Rate (Ci/sec) | µCi's Released | * EA(610m) Concentration(µCi/ml) ( $\chi/Q$=1.3E-3s/m$^3$) | 10CFR20 Table II Col. 1 MPC | $\frac{\chi}{MPC}$ |
|---|---|---|---|---|---|
| Cs-137 | 6.01E-8 | 9.0E-1 | 7.8E-11 | 5E-10 | 0.156 |
| Cs-134 | 4.03E-9 | 6.0E-2 | 5.2E-12 | 4E-10 | 0.013 |
| Sr/Y-90(EI)[t] | 2.79E-9 | 4.2E-2 | 3.6E-12 | 3E-11 | 0.12 |
| Ru-106 | 3.4E-11 | 5.1E-4 | 4.4E-14 | 2E-10 | 2.2E-4 |
| Ce-144 | 3.3E-11 | 5.0E-4 | 4.3E-14 | 2E-10 | 2.2E-4 |
| Sb-125 | 1.97E-10 | 3.0E-3 | 2.6E-13 | 9E-10 | 2.9E-4 |
| Te-125m | 4.7E-11 | 7.1E-4 | 6.1E-14 | 4E-9 | 1.5E-5 |
| I-129 | 4.3E-14 | 6.5E-7 | 5.6E-17 | 2E-11 | 2.8E-6 |
| Th-234 | 1.0E-15 | 1.5E-8 | 1.3E-18 | 1E-9 | 1.3E-9 |
| Pa-233 | 4.5E-17 | 6.8E-10 | 5.9E-20 | 6E-9 | 9.8E-12 |
| U-234 | 4.5E-15 | 6.8E-8 | 5.9E-18 | 4E-12 | 1.5E-6 |
| U-235 | 1.6E-16 | 2.4E-9 | 2.1E-19 | 4E-12 | 5.3E-8 |
| U-236 | 1.4E-16 | 2.1E-9 | 1.8E-19 | 4E-12 | 4.5E-8 |
| U-237 | 1.4E-16 | 2.1E-9 | 1.8E-19 | 1E-10 | 1.8E-9 |
| U-238 | 1.0E-15 | 1.5E-8 | 1.3E-18 | 3E-12 | 4.3E-7 |
| Np-237 | 4.5E-17 | 6.8E-10 | 5.9E-20 | 1E-13 | 5.9E-7 |
| Np-239 | 6.4E-18 | 9.6E-11 | 8.3E-21 | 2E-8 | 4.2E-13 |
| Pu-236 | 2.5E-18 | 3.8E-11 | 3.3E-21 | 2E-14 | 1.7E-7 |
| Pu-238 | < 5.7E-14 | < 8.6E-7 | < 7.4E-17 | 7E-14 | < 1.06E-3 |
| Pu-239 | 3.8E-13 | 5.7E-6 | 4.9E-16 | 6E-14 | 8.2E-3 |
| Pu-240 | 1.2E-13 | 1.8E-6 | 1.6E-16 | 6E-14 | 2.7E-3 |
| Pu-241 | 9.3E-12 | 1.4E-4 | 1.2E-14 | 3E-12 | 4.0E-3 |
| Pu-242 | < 1.3E-17 | < 2.0E-10 | < 1.7E-20 | 6E-14 | < 2.8E-7 |
| Am-241 | 5.3E-14 | 8.0E-7 | 6.9E-17 | 2E-13 | 3.5E-4 |
| Am-243 | 6.4E-18 | 9.6E-11 | 8.3E-21 | 2E-13 | 4.2E-8 |

$\frac{\chi}{MPC}$ =0.306

* EA=Exclusion Area; EI=each isotope

In the determination of the "exclusion area" concentration, the value for $\frac{\chi}{Q}$ of 1.3 E-3 s/M$^3$ was used because of the short duration of the release which precluded the use of the annual average $\frac{\chi}{Q}$. Use of the chosen $\frac{\chi}{Q}$ yields an EA concentration 570 times the concentration calculated using the highest sector (SE) annual average $\frac{\chi}{Q}$ of 2.27E-6 sec/M$^3$.

As shown at the bottom of column 6, the summation of the individual $\frac{\chi}{MPC}$ is less than 1.0, i.e. 0.306, for this pessemistically conservative scenario.

Factors of conservatism are as described below:

1) credit has been taken only for 1 bank of HEPA filters; the effluent has to pass through 2 banks of HEPA filters before being released to the atmosphere

2) the $\frac{X}{Q}$ values used for the calculation is a plume centerline value and is a value higher than would be experienced at ground level at the 610 meter distance.

In addition to the comparison with 10CFR20 off-site MPC's, doses were calculated for the inhalation pathway for two individuals assumed to be located at the exclusion area distance of 610 meters at the plume centerline forthe full duration of the plume passage. The dose was calculated for the seven radiologically significant isotopes for an infant and an adult to give an indication of the magnitude of the doses. Values are as indicated in the tables below. Breathing rates for the two individuals were calculated from Reg. Guide 1.109 by dividing the annual rates by $3.15 \times 10^7$ sec/yr. Total volumes breathed during the plume passage are then the respective breathing rate ($M^3$/sec) times 15 seconds, the duration of the release. Dose conversion factors for Cs-137, Cs-134 and Sr-90 for the respective individuals are as tabulated in the Off-site Dose Calculation Manual, Rev. 4. Doses for the plutonium isotopes (Pu-238, Pu-239, & Pu-241) and americium-241 were calculated according to the ICRP Task Group on Lung Dynamics as outlined in Ref. 1. Particle size for the inhaled plutonium particles was assumed to be 0.3 microns, a size which could theoritically pass through the HEPA filters.

Adult Doses

| Isotope | pCi Inhaled | Doses (mrem) for the listed organ: Bone | Liver | T. Body | Kidney | Lung | GI-LLI |
|---|---|---|---|---|---|---|---|
| Cs-137 | 2.96E-1 | 1.8E-5 | 2.3E-5 | 1.6E-5 | 8.2E-6 | 2.8E-6 | 3.1E-7 |
| Cs-134 | 1.98E-2 | 9.2E-7 | 2.1E-6 | 1.8E-6 | 7.1E-7 | 2.4E-7 | 2.6E-8 |
| Sr-90 | 1.37E-2 | 1.7E-4 | - | 1.0E-5 | - | 1.6E-5 | 1.2E-6 |
| Pu-238 | 2.8E-7 | 9.6E-7 | 1.4E-6 | - | - | 1.1E-7 | - |
| Pu-239 | 1.86E-6 | 5.9E-6 | 8.5E-6 | - | - | 6.9E-7 | - |
| Pu-241 | 4.6E-5 | 1.5E-7 | 2.1E-7 | - | - | 1.7E-8 | - |
| Am-241 | 2.6E-7 | 8.9E-7 | 1.3E-6 | - | - | 1.0E-7 | - |
| Total Organ Dose | | 1.97E-4 | 3.7E-5 | 2.8E-5 | 8.9E-6 | 2.0E-5 | 1.5E-6 |

Infant Doses

| Isotope | pCi Inhaled | Doses (mrem) for the listed organ: Bone | Liver | T. Body | Kidney | Lung | GI-LLI |
|---|---|---|---|---|---|---|---|
| Cs-137 | 5.2E-2 | 2.0E-5 | 2.3E-5 | 1.7E-6 | 6.4E-6 | 2.6E-6 | 5.0E-8 |
| Cs-134 | 3.5E-3 | 9.9E-7 | 1.8E-6 | 1.9E-7 | 4.8E-7 | 2.0E-7 | 3.3E-9 |
| Sr-90 | 2.4E-3 | 7.0E-5 | - | 4.4E-6 | - | 1.9E-5 | 2.2E-7 |
| Pu-238 | 4.9E-8 | 4.0E-7 | 6.2E-7 | - | - | 1.3E-7 | - |
| Pu-239 | 3.3E-7 | 2.5E-6 | 3.9E-6 | - | - | 8.1E-7 | - |
| Pu-241 | 8.1E-6 | 6.2E-8 | 9.6E-8 | - | - | 2.0E-8 | - |
| Am-241 | 4.6E-8 | 3.8E-7 | 5.8E-7 | - | - | 1.2E-7 | - |
| Total | Organ Dose | 9.4E-5 | 3.0E-5 | 6.3E-6 | 6.9E-6 | 2.3E-5 | 2.7E-7 |

In all cases where plutonium or americium is concerned, the particle size was assumed to be 0.3μ and the retention factor was assumed to be 30%. Also, all doses are time integrated to give dose commitments to 70 years of age. For the adult, assumed to be 20 years of age, the dose is a 50-year dose commitment and for the infant it is a 70 year dose commitment.

In either case, although the microcuries of the cesiums and strontium inhaled are orders of magnitude greater than for the plutonium and americium isotopes, the higher dose conversion factors for Pu + Am create a dose roughly 10% of those due to Cs + Sr. In any case, although the dose has not been calculated for each isotope, the dose was calculated for 7 of the radiologically dominant isotopes. As shown for the tabulated total organ doses, the highest is the adult "bone" dose of $2.0 \times 10^{-4}$ mrem. Therefore, all doses calculated for the exclusion area are within acceptable limits. Doses beyond this location would be lower than the tabulated values.

Therefore, as determined under the assumptions of this analysis, the significant radiological concerns for such a hypothetical demineralizer spill involve the personnel doses incurred during cleanup. Off-site exposures for such an event are relatively insignificant. Unless a demineralizer tank rupture is determined to be a significant possibility, radiological concerns are not a driving force in the disposition of the MU&P demineralizer contents.

Due to the higher transuranic and Sr-90 inventory of the "A" demineralizer, off-site inhalation doses have been calculated assuming that the MHA involves the "A" demineralizer. Doses are tabulated below. Even though these doses are slightly higher than those potentially resulting from the "B" demineralizer, they are still within acceptable limits.

Adult Doses

| Isotope | pCi Inhaled | Doses (mPem) for the listed organ: Bone | Liver | T. Body | Kidney | Lung | GI-LLI |
|---|---|---|---|---|---|---|---|
| Cs-137 | 1.55E-1 | 9.4E-6 | 1.2E-5 | 8.4E-6 | 4.3E-6 | 1.5E-6 | 1.6E-7 |
| Cs-134 | 1.04E-2 | 4.8E-7 | 1.1E-6 | 9.4E-6 | 3.7E-7 | 1.3E-7 | 1.4E-8 |
| Sr-90 | 1.31E-1 | 1.6E-3 | - | 9.5E-5 | - | 1.5E-4 | 1.1E-5 |
| Pu-238 | 1.40E-6 | 4.8E-6 | 7.0E-6 | - | - | 5.5E-7 | - |
| Pu-239 | 9.30E-6 | 2.9E-5 | 4.3E-5 | - | - | 3.5E-6 | - |
| Pu-241 | 2.30E-4 | 7.5E-7 | 1.1E-6 | - | - | 8.5E-8 | - |
| Am-241 | 1.30E-6 | 4.5E-6 | 6.5E-6 | - | - | 5.0E-7 | - |
| Total | Organ Dose | 1.65E-3 | 7.07E-5 | 1.14E-4 | 4.67E-6 | 1.56E-4 | 1.12E-5 |

## Infant Doses

| Isotope | pCi Inhalded | Doses (mRem) for the listed organ: Bone | Liver | T. Body | Kidney | Lung | GI-LLI |
|---|---|---|---|---|---|---|---|
| Cs-137 | 2.7E-2 | 1.0E-5 | 1.2E-5 | 8.9E-7 | 3.3E-6 | 1.4E-6 | 2.6E-8 |
| Cs-134 | 1.8E-3 | 5.2E-7 | 9.4E-7 | 9.9E-8 | 2.5E-7 | 1.0E-7 | 1.7E-9 |
| Sr-90 | 2.3E-2 | 6.7E-4 | - | 4.2E-5 | - | 1.8E-4 | 2.1E-6 |
| Pu-238 | 2.5E-7 | 2.0E-6 | 3.1E-6 | - | - | 6.5E-7 | - |
| Pu-239 | 1.7E-6 | 1.3E-5 | 2.0E-5 | - | - | 4.1E-6 | - |
| Pu-241 | 4.1E-5 | 3.1E-7 | 4.8E-7 | - | - | 1.0E-7 | - |
| Am-241 | 2.3E-7 | 1.9E-6 | 2.9E-6 | - | - | 6.0E-7 | - |
| Total Organ Dose | | 7.0E-4 | 3.9E-5 | 4.3E-5 | 3.6E-6 | 1.9E-4 | 2.1E-6 |

## Reference

1 - "Calculation of Doses Due to Accidently Released Plutonium from an LMFBE", by R. R. Fish, et. al., November 1972, (ORNL-NSIC-74).