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TMI-2 Cleanup Project Directorate
Attn: Dr. W. D. Travers
Director
US Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, PA 17057

Dear Dr. Travers:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
TMI-2 Temporary Reactor Vessel Filtration System
Safety Evaluation Report - Revision 1

Attached is Revision 1 to the TMI-2 Temporary Reactor Vessel Filtration System (TRVFS) Safety Evaluation Report (SER) for your information. This revision reflects revisions to the calculations relating dose rates to fuel content of the TRVFS. As previously discussed with you and members of your staff, we have reviewed our previous calculations and have made revisions based on a more realistic but conservative approach; the changes are summarized below.

Upon re-evaluation, it was decided to revise the focal point to the calculations to use the discharge barrel as the limiting feature (70 Kg) and establish a limit of 10 discharges to the barrel from the filter. This results in a limit of fuel in the filter of 7 Kg. Dose rates were then calculated based on the 7 Kg loading in the filter.

The revised calculations giving radiation levels at the filter housing included the following parameter assumptions:

- o The new analysis used 7 Kg of $UO_2$ versus 70 Kg of fuel as used in the previous analyses.
- o The new analysis uses a source region material mix of $UO_2$ fines, structural material fines, diatomaceous earth and water versus all $UO_2$.

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- o The source region density is assumed to be 1.3 gms/cc versus 4 gm/cc.
- o The new analysis uses a core average neutron flux versus a hot channel factor for determining specific activity of the core debris.
- o The radiation detector distance to the filter housing is assumed to be two (2) inches instead of one (1) inch and is located at the centerline of the filter region as opposed to bottom of the filter region.
- o The fission product activity is decayed to October 1985 versus February 1985 in the previous analysis.

The resulting dose rate at the filter housing using the above assumptions is approximately 340 R/hr. Assuming a radiation alarm setpoint of 3 R/hr on the filter housing monitor, the result is a safety margin of approximately 110. If a setpoint of 6 R/hr is used, the safety factor becomes approximately 55. If the soluble fission products (Cs 134, Cs 137, Co 60, and Sb 125) are assumed depleted from the core debris, the radiation level at the filter housing is 128 R/hr, yielding safety factors of approximately 40 and 20 for alarm setpoint of 3 and 6 R/hr respectively.

Possible increase of the alarm setpoint is under consideration as an ALARA measure since the filter is currently being changed out based on radiation levels as opposed to ΔP. The frequent changes are adding significantly to the dose rates for defueling operations. Appropriate changes may be made in order to limit personnel doses based on operating experiences.

As you are aware, we suspended operation of the TRVFS pending completion of our re-evaluation. Based on the results of our evaluations presented above and in Revision 1 to the TRVFS SER, GPU Nuclear will resume operation of this temporary system while a permanent system is being developed. This is as discussed previously with yourself and members of your staff.

Sincerely,

F. R. Standerfer
Vice President/Director, TMI-2

FRS/RBS/eml

GPU Nuclear

# SAFETY ANALYSIS

SA # 4340-3220-86-0023

Rev. # 1



## TITLE

TMI-2 Temporary Reactor Vessel Filtration System

Originator Richard E. Sheppard Date 3/5/86

CONCURRENCE

Lead Engineer/RTR R. B. Swartzwelder Date 3/5/86 | SRG [signature] Date 3/5/86

Cognizant Eng. JO Lochabill Date 3/5/86 | Rad Con [signature] Date 3/5/86

APPROVAL

Mgr. Eng. Services [signature] Date 3-5-86 | Site Ops Director [signature] Date 3/5/86

Mgr. Recovery Pgms. [signature] Date 3/5/86

# TMI-2 TEMPORARY REACTOR VESSEL FILTRATION SYSTEM

## 1. Purpose and Scope

The purpose of this Safety Analysis is to demonstrate that the operation of the Temporary Reactor Vessel Filtration System will not present an undue risk to the health and safety of the public.

The prime purpose of the Temporary Reactor Vessel Filtration System is to restore and maintain the visibility in the Reactor Vessel to acceptable levels to insure the continuation of the Early Defueling Program. Recent developments relative to the operation of the DWCS and the Filter Canisters have revealed that the Filter Canisters develop the maximum design pressure drop before the filter has processed significant quantities of water. Our investigation of this development has lead to the discovery of micro-organism growth in the reactor coolant. Theory and experience indicates that these micro-organisms are capable of plugging the filters in the filter canister prior to the collection of any significant quantity of core debris. These developments have provided the need to design and operate a temporary filter system while GPU Nuclear develops a permanent program to control this phenomenon.

The principle safety questions relating to the system's operation are criticality control, waste disposal, and the potential consequences of spills.

## 2. System Description

The Temporary Reactor Vessel Filtration System (TRVFS) is provided to cleanup the Reactor Vessel water above the rubble bed to provide and maintain an adequate level of visibility to enable defueling to proceed. The system consists of a 1 1/2 hp pump, 1 1/2 inch diameter hoses, isolation valves, fittings and a filter assembly. The TRVFS will be operated only when operation personnel are on the defueling platform. The TRVFS will take suction from the IIF and return the filtered water to the IIF. Flow will be provided by the pump at maximum flow rate of 75 gallons per minute. The water will pass through a diatomaceous earth (D.E.) coated filter. The filter is housed in a container with filtration provided by approximately 100 15 inch long by 1/2 inch diameter filter bags on which the diatomaceous earth precoat is fixed. The diatomaceous earth is injected into the suction of the pump in approximately seven (7) pound batches which then coats the filter medium. When the pressure drop across the filter reaches 8-10 psi above precoat pressure drop, the pump will no longer be capable of providing significant flow; consequently, the filter will be "backbumped" or cleaned. Backbumping is accomplished by stopping the flow and flexing the filter media which causes the filtered material and the diatomaceous earth to fall to the bottom of the filter housing. The bottom of the housing can be drained to a shielded container such as a 55 gallon drum as required. The TRVFS is restored by establishing flow and injecting an additional seven (7) pounds of clean diatomaceous earth into the pump suction. To assure that filtered material in the filter is not pumped back to the reactor vessel before the recharged D.E. is fixed to the filter, a bypass line is provided to allow recirculation until the effluent from the filter appears clear. Each cleaning of the filter will add approximately 7 to 20 pounds of diatomaceous earth, filtered material, and 20-25 gallons of water. After settling has occurred in the shielded container, the water will be decanted and stored for further processing at a later date.

## 3. Criticality Prevention

Any fluid system connected to the vessel which transports coolant system water from the IIF has the potential to move fuel bearing material. Consequently, the potential to accumulate fuel external of the reactor vessel has been addressed. Because of the temporary nature of the system and the unlikelihood of accumulating significant quantities of fuel based on the suction point for this system, GPU Nuclear believes that the rigorous engineering/design control established for defueling canisters, which were designed for extended life, is not required in this instance. However, the TRVFS design and operation does provide the following separate assurances to preclude any potential significant fuel accumulation and criticality.

a. Only suspended material in the RCS will be moved.

b. The filter body will be continuously monitored for gamma radiation levels to detect any significant fuel accumulation.

c. The gross alpha radioactivity in the 55 gallon drum will be determined by sampling.

The inlet and outlet hoses for the TRVFS are one and one-half (1 1/2) inch I.D. hoses. The hoses from the IIF to the pump and from filter to IIF will be secured in such a manner that the suction and discharge piping will always be immersed in the IIF no more than two (2) feet below 327'-6". Consequently, the suction of the RCS water will always occur within the confines of the IIF. At 75 gpm, the velocity in the hose is approximately 14 feet per second; however, the fluid velocity ten feet from the hose would be four to five orders of magnitude less, if any velocity effect from the hose suction exists at all. At such velocities only particles smaller than 10 microns could be moved. It could be assumed that the suspended material in the reactor water is uranium oxide at a concentration of 1 ppm; representative of what sample concentration have indicated in the past. Twelve hours of continuous operation of the filter system at this concentration would deposit 0.2 Kg of $UO_2$ on the filter media, significantly less than that required to produce a criticality. The conservative nature of this hypothetical model is illustrated by comparison to the analysis of the pre and post-filter effluent in the DWCS operation; a similar but deeper suction and discharge arrangement. Analysis of DWCS fluid has shown no detectable fissile material in the flow stream. Therefore, it is concluded that a significant accumulation of uranium oxide will not occur in the filter housing.

A gamma sensitive radiation detector is provided in close proximity to the filter housing. Calculations using upper limit radionuclide removal rates based on scaling empirical data from DWCS and RCS samples indicate that after filtering RCS water for four (4) hours the highest expected radiation level at the exterior surface of the vessel would be approximately 3 R/hr. If, however, the vessel contained 7 Kg of $UO_2$, with the attendant fission product activity as predicted by the ORIGEN code, the radiation level at the container surface would be 340 R/hr. This significant difference provides an excellent check relative to any possible uranium accumulation in the filter during operation. 7 Kg of fuel as a maximum filter loading was determined by taking the minimum amount of fuel (70 Kg) necessary for criticality concerns in the drum, then dividing the 70 Kg by the maximum allowed discharges (10) to the drum. The radiation levels in the filter housing was then determined using the following assumptions:

- o The source region material includes $UO_2$ fines, structural material fines, diatomaceous earth, and water.
- o Source region density of 1.3 gm/cc
- o Core average neutron flux distribution for specific activity of debris
- o Source to detector distance of approximately 2 inches at the centerline of filter media region
- o Activity decayed to October 1985

A gamma radiation monitor will continuously monitor the surface radiation level and at a preselected radiation level filtering operations will be secured. Currently this level is procedurally controlled with values expected to be up to 6 R/hr. This level may be adjusted by Rad Con to account for changing conditions. At this radiation level, the amount of $UO_2$ in the filters is expected to be gram quantities, far less than necessary to create criticality concerns.

As an additional check to assure that little or no fissile material and/or fuel bearing material is being removed from the RCS, grab samples will be periodically taken of the diatomaceous earth and filtered material in the 55 gallon drum. A test for gross alpha radioactivity will be used to calculate the quantity of fissile material in the drum. The drum may be used to accumulate as many as ten (10) cleanings of the filter media. Since little or no fuel is expected in any filter loading, it is obvious that insufficient fuel could accumulate in the drum to create criticality concerns.

The potential for criticality due to a boron dilution event has been considered. Diatomaceous earth consists of approximately 88% silica and exhibits no propensity to remove or adsorb boron. Therefore, significant boron dilution caused by removal of boron by the diatomaceous filters is not considered credible. Operating experience with these filters in the fuel pool has resulted in no detectable dilution of the fuel pool. Boron dilution of the reactor vessel is not judged credible because of the closed loop nature of the system, and the unavailability of unborated water sources.

4. Waste Disposal

Calculations of the estimated radionuclide concentrations and the maximum expected concentrations of fissile material indicate that the raw waste is slightly greater than those required for disposal as a Class C package. The group 2 long-lived isotopes are controlling for Class C. Therefore, stabilization will be required for shallow land burial. Cement solidification of the waste will reduce the concentrations to those acceptable for shipment as a Class C waste.

Consequently, it is concluded that shipment of these wastes will not represent an abnormal waste disposal concern for this program.

5. Dose Rate Evaluation and Spill Consequences

As stated previously, the expected maximum radiation level at the external surface of the filter housing is 6 R/hr. Shadow shielding will be installed so that the contribution that this localized radiation source will make to the exposure of operating personnel on the Defueling Platform will be minimal. Radiation levels from the shielded 55 gallon drum are expected to be as much as 340 mR on contact. Shadow shielding will be used to reduce the dose rate from this source to personnel working at the defueling slot to one (1) to two (2) mR/hr. It may be possible during the transfer of D.E. from filter to 55 gallon drum to experience a spill wherein the 20 to 25 gallons of water and the approximately seven (7) pounds of D.E. along with the filtrate are spilled onto the surface of the North End Defueling Platform. Should such a spill occur, a portion of the platform would be contaminated with up to seven (7) curies of Strontium/Yttrium-90 and 0.07 curies of Cesium-137. If the spill spreads to cover a depth of 1/8" (3 mm), an area of about 200 $ft^2$ will be contaminated. Dose rates attributable to this contamination will be in the range of 25 rad per hour at 10 cm above the floor.

Although a specific calculation for offsite release created by a drum handling accident has not been performed, the results are easily bounded by and are far less than the releases postulated for a canister drop accident as described in Revision 4 to the Early Defueling SER.

A liquid only spill must also be considered. A pipe break at the pump discharge would exhibit the potential of spilling liquid from the IIF onto the 322'-6" elevation of the FTC floor. This event can be detected using the IIF level monitoring system. This liquid would in turn drain to the sump of the canal floor on the south-east corner of the upper canal where it would collect and be pumped to a staging or processing location. With the suction limited to two (2) feet below the surface of the water in the IIF, this represents approximately 4000 gallons of RCS water. It is not expected that such an event would significantly increase the radiation exposure to workers on the Defueling Platform.

Another potential concern relates to the consequences of a filter break through wherein the seven (7) pounds of diatomaceous earth is pumped into the reactor vessel. As previously stated, diatomaceous earth is chiefly silica with little, if any, hydrogeneous material, in a fine powder form. It would, therefore, significantly increase the turbidity of the RCS fluid but not effect the present shutdown margin of the bulk core. Eventually, it would be expected to settle to the top of the rubble bed and in turn be removed with the rubble during defueling.

## 6. Summary

Therefore, it is concluded that based on the evaluations presented in this Safety Analysis the operation of the Temporary Reactor Vessel Filtration System may be conducted without undue risk and exposure to the operating personnel nor will it present any undue risk to the health and safety of the public.

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, text, or experiment is determined not to be an unreviewed safety question and does not involve a modification to 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 analysis may be increased; or

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

c. The margin of safety, as defined in the basis for any Technical Specifications, is reduced.

A variety of events have been analyzed in this SER. It has been determined that the event due to operation of the TRVFS are similar to events described in several previous submittals (References 1, 2 and 3). Of primary concern are a handling accident, deboration of the RV, draindown of the RV and criticality concerns in the filter. The drop of the 55 gallon container and subsequent release to the environment is bounded by the canister drop accident described in Reference 1. Deboration of the RV is possible by one of the two methods: absorption of boron by the D.E. or by dilution caused by improper hookup and operation of the filter system. The filter media has shown no propensity to remove boron during test operations in the fuel pool. Operation of similar types of systems have been addressed in the Boron Hazards Analysis (Reference 2) and are not considered to be a significant dilution hazard.

Due to the setup of hose suction and discharges, draindown of the reactor vessel is not considered credible. Previous evaluations have shown that ambient cooling is adequate providing water remains above the vessel flange. Since draindown will be limited to the upper two (2) feet of the IIF, this event is bounded.

It is recognized that the criticality control measures for the filter canister proposal here does not incorporate the engineered features for criticality control used in the defueling filter canisters (Reference 3). However, measures have been taken to minimize and detect the buildup of $UO_2$ in the filter media. These effects, coupled with the temporary nature of the proposed system, give confidence that the proposed operation does not increase the probability of an accident or create the possibility of an accident of a different type than previously evaluated.

Technical Specification safety margins at TMI-2 are concerned with criticality controls and prevention of further core damage due to overheating. As demonstrated by this Safety Evaluation Report, Technical Specification safety margins will be maintained throughout the filtering process. Subcriticality is ensured by establishing the boron concentration at greater than 4350 ppm during the early defueling process and ensuring that this concentration is maintained by monitoring the boron concentration and inventory levels and by isolating potential deboration pathways. Subcriticality in the filter system is maintained by selective pickup locations, monitoring of radiation levels and by sampling of the discharged filter media. The ability to prevent further core damage due to overheating is not affected by the filtering process.

Thus, it is concluded that the operation of the Temporary Reactor Vessel Filtration System does not constitute an unreviewed safety question as defined by 10 CFR Part 50, Paragraph 50.59.

## REFERENCES


1. Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel, Revision 4, GPU Nuclear letter 4410-85-L-0200, dated October 10, 1985, from F. R. Standerfer to B. J. Snyder.

2. Boron Hazards Analysis, Revision 2, GPU Nuclear letter 4410-85-L-0195, dated September 27, 1985, from F. R. Standerfer to B. J. Snyder.

3. Technical Evaluation Report for Defueling Canisters, Revision 1, GPU Nuclear letter 4410-85-L-0183, dated September 10, 1985, from F. R. Standerfer to B. J. Snyder.