Met-Ed / GPU

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Writer's Direct Dial Number

December 15, 1981
LL2-81-0286

TMI Program Office
Attn: Dr. B. J. Snyder, Program Director
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety and Radiological Evaluation of the
Containment Building Decontamination Experiment

Reference: Letter J. J. Barton to Dr. B. J. Snyder, dated November 18, 1981.

Attached is the revised Radiological and Safety Evaluation for the proposed Decontamination Experiment. This document is being submitted for your review and approval.

This document was revised based on the results of the November 25, 1981 meeting at which you and your staff were present. GPU has included in this revised document the agreed upon resolutions to the concerns which were expressed by you and your staff at that meeting.

If you have any further concerns or questions, please call J. E. Larson of my staff on (717) 948-8832.

Sincerely,

J. J. Barton
Acting Director, TMI-2

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JJB:SWS:djb

Attachment

cc: L. H. Barrett, Deputy Program Director, TMI Program Office

RADIOLOGICAL AND SAFETY EVALUATION OF THE CONTAINMENT BUILDING DECONTAMINATION EXPERIMENT FOR TMI-2 RECOVERY

Revision 1
December, 1981

1.0 SCOPE

The information in this document represents the radiological and safety evaluation of activities to be performed in the containment as given in the "Containment Building Decontamination Experiment Purpose and Scope of Work" transmitted to Dr. B. J. Snyder on October 30, 1981. Included in the evaluation are the following:

a) Effluents to the environment
b) Occupational exposures.
c) Radioactive waste management.
d) Industrial Safety
e) Safety evaluation (10 CFR 50.59).

2.0 OFFSITE RADIOLOGICAL DOSES

2.1 Effluents to the environment

During the performance of the activities identified in the scope document referenced in Section 1, there will be two types of airborne radioactivity in the containment which are considered in the safety evaluation of effluents to the environment:

1) Particulate

2) Tritium

All other airborne activity is considered negligible.

A portion of the assumed airborne particulate activity in the containment will be exhausted to the environment through the contaiment ventilation system which contains High Efficiency Particulate Adsorber (HEPA) filters. Tritium concentrations in the containment have been measured, from which total tritium quantities were calculated. This quantity of tritium is assumed to be released to the environment through the ventilation system. The discussions of how the source terms for the particulate activity and tritium activity were developed are presented in sections 2.1.1 and 2.1.2 respectively.

Two conservative assumptions which were used in the calculations of effluents should be mentioned here. They are:

1) It is assumed that the containment purge will be operated continuously for a 13 week period. This is considered conservative because the activities associated with the decontamination experiment will be completed in a 6 to 8 week period and the purge will not be operated continuously.

2) It is assumed the containment purge will be operated at 50,000 CFM, However, when based on past experience, the actual purge rate will be less than 25,000 CFM.

2.1.1 Particulate Releases

In order to calculate the airborne particulates released to the environment, a source term was determined. Three sources of data were considered in the determination of the airborne particulate source term.

1) HPR-227
2) Penetration R-626
3) HPR-219A

Measurements of airborne activity in the containment atmosphere, as measured by HPR-227, were reviewed and an average particulate airborne concentration of 3×10^{-10} μCi/cc for Cs-137 approximated. Concentrations of other isotopes were calculated by assuming the total radioactive particulates consist of 84 percent Cs-137, 12 percent Cs-134, and 4 percent Sr-90. This is consistent with the relative concentrations found in the containment surface samples. The assumed concentration for the various isotopes are given in Table 2-1.

The use of HPR-227 as the source of data for the determination of source terms is considered appropriate based on the information provided in GEND 009, "Measurements of ^{129}I and Radioactive Particulate Concentrations in the TMI-2 Containment Atmosphere During and After the Venting". This report contains the results of direct measurements of the containment airborne particulate activity taken through penetration R-626. A comparison of the R-626 data with the results of HPR-227 particulate samples taken during the same time period indicates that HPR-227 yields slightly higher concentrations than does R-626. Therefore, using the data obtained from HPR-227 is considered appropriate for approximating the airborne particulate concentration in the containment atmosphere.

Also, it was assumed the particulate airborne concentration as given in Table 2-1 remains constant throughout the assumed 13 week period. This means it was assumed that there is no increase in airborne particulate activity as a result of activities being performed in the containment. This is considered acceptable based on a review of operating experience gained during previous entries. Specifically, the plant vent stack releases since July 1980, as measured by HPR-219A, have shown no detectable increase in particulate activity released to the environment during times when activities were occurring in the containment, including during the decontamination test performed in May 1980. This supports the assumption that the gross containment particulate airborne concentration remains essentially constant, for the planned activities occurring in the containment.

Using the assumptions described above and a HEPA filter efficiency of 99.9 percent, the quantities of radioactive particulates which may be released to the environment were calculated. The results of these calculations are given in Table 2-2.

Using the results given in Table 2-2, the resulting doses to individuals were calculated in accordance with the guidance provided in Regulatory Guide 1.109. The doses were calculated at the following locations:

a) nearest residence (0.5 mile, E)

b) nearest garden (1.1 mile, ENE)

c) nearest milk goat (1.2 mile, N)

d) nearest milk cow and garden (1.1 mile, E)

The calculated doses are given in Table 2-3.

An additional analysis was performed for the nearest garden. This analysis used the meteorological data (X/Q and D/Q) presented in the Final Programmatic Environmental Impact Statement, NUREG-0683, Appendix W, instead of the meteorological data from the TMI Offsite Dose Calculation Manual. The source terms for this analysis are given in Table 2-2. The bone dose for the child is given in Table 2-5.

2.1.2 Tritium Release

Measurements of the tritium concentration in the containment atmosphere, as measured at the HPR-227 Sampling Station, were reviewed and an average airborne concentration of 10^{-6} μCi/cc was approximated. Using this value for the source term, the quantity of tritium released to the environment was calculated based on the following assumptions:

a) The tritium concentration of 10^{-6} μCi/cc remains constant for the 13 week duration.

b) The containment purge exhaust is operated continuously for the 13 week duration at a flowrate of 50,000 cfm.

The quantity of tritium released was calculated to be 186 Ci. Using the value of 186 Ci and the guidance provided in Regulatory Guide 1.109, the resulting doses to individuals were calculated. These results are presented in Table 2-4.

2.1.3 Discussion of Results

If the offsite doses given in Tables 2-3 and 2-4 are compared to the limits given in appendix B, section 2.1 of the TMI-2 Technical Specifications, it can be seen they are a small fraction of the specified limits. Although the calculated doses are only from in-containment sources, by comparing calculated releases to measured stack releases, it can be concluded these calculations are enveloping for the decontamination experiment. This is based on data accumulated to date which indicates that there were no detectable increases in measured stack releases (as measured by HPR-219A) when comparing periods when the containment purge was operating to periods when it was not. This means that when activities were being conducted in the containment similar to those planned for the decontamination experiment, no detectable releases were measured which could be directly attributed to those activities. Also, even if the calculated release for in-containment work associated with the decontamination experiment is added to the average measured stack release to date, the resultant release would still be well within the technical specification limits. Therefore, it is reasonable to expect the decontamination experiment will not result in any measurable increase in releases or offsite doses.

TABLE 2-1

AVERAGE PARTICULATE AIRBORNE RADIOACTIVITY
CONCENTRATIONS IN THE CONTAINMENT (NOTE 1)

Radionuclide	Concentration (μCi/cc)
Cs-134	4.3×10^{-11}
Cs-137	3.0×10^{-10}
Sr-90	1.4×10^{-11}

NOTE 1:

Each particulate airborne sample that is sent to the Sample Coordinator for analysis is typically counted on the 39 percent efficient Ge(Li) gamma spectrometer for 1000 seconds. Each peak in the resulting spectrum is then compared to a list of 43 radionuclides in the computer library. Only positive identifications are then entered on the Radio-Chemistry Analysis Summary Sheet for that particular sample. LLD's for the other nuclides will be known but not reported.

(NOTE 1 CONTINUED ON NEXT PAGE)

The analysis of a typical air sample; i.e., 4 CFM for 15 minutes, will attain the following LLD's for a 1000 second count:

Cr-51	2.3 E-10 μCi/cc
Mn-54	1.4 E-11 μCi/cc
Fe-59	2.9 E-11 μCi/cc
Co-58	1.5 E-11 μCi/cc
Co-60	1.4 E-11 μCi/cc
Zn-65	3.6 E-11 μCi/cc
Ag-110m	1.1 E-10 μCi/cc
Zr-95	2.4 E-11 μCi/cc
Nb-95	1.6 E-11 μCi/cc
Mo-99	1.6 E-11 μCi/cc
Tc-99m	1.9 E-11 μCi/cc
Ru-103	3.3 E-11 μCi/cc
Ru-106	2.5 E-10 μCi/cc
Sn-113	4.9 E-11 μCi/cc
Cs-136	1.2 E-11 μCi/cc
Ce-141	3.3 E-11 μCi/cc
Ba-140	7.5 E-11 μCi/cc
La-140	1.2 E-11 μCi/cc
Sb-125	1.1 E-10 μCi/cc
Ce-144	1.4 E-10 μCi/cc

All these LLD's are less than the MPC's for unrestricted exposure. Air samples taken within the Reactor Building typically do not detect these nuclides, and after passing through the HEPA filters in the purge exhaust train the possiblity of seeing these nuclides in the stack effluent is even further reduced. In general terms, the two (2) gamma emitters of abundance in the Reactor Building are Cs-134 and Cs-137.

(NOTE 1 CONINUED ON NEXT PAGE)

A gross alpha count is also performed if requested. If the result is positive, the sample is held for 72 hours to allow for decay of naturally occurring radionuclides and then recounted. When recounted, the air samples typically indicate LLD of 2.5 E-13 μCi/cc. This LLD is less than the restricted area MPC (6 E-13 μCi/cc) which must be used when unknown alpha emitters are present.

TABLE 2-2

CALCULATED PARTICULATE AIRBORNE RELEASES TO THE ENVIRONMENT
(13 WEEK CONTINUOUS PURGE)

Radionuclide	Release (Ci)
Cs-134	8.0×10^{-6}
Cs-137	5.6×10^{-5}
Sr-90	2.6×10^{-6}

TABLE 2-3

DOSE TO INDIVIDUALS FROM ALL PATHWAYS
FOR PARTICULATE RELEASES
(13 WEEK CONTINUOUS PURGE)

Location/Pathway	Dose (mrem)				
	Total Body	Bone	Lung	Skin	Liver
Nearest Residence[a]					
Inhalation	1.9×10^{-6}	1.2×10^{-5}	1.2×10^{-6}	--------	1.8×10^{-6}
Ground Plane	6.4×10^{-5}	---------	--------	7.5×10^{-5}	---------
Total	6.6×10^{-5}	1.2×10^{-5}	1.2×10^{-6}	7.5×10^{-5}	1.8×10^{-6}
Nearest Garden[b]					
Inhalation	7.1×10^{-7}	8.8×10^{-6}	1.2×10^{-6}	--------	1.5×10^{-6}
Vegetable	2.4×10^{-4}	1.0×10^{-3}	4.0×10^{-5}	--------	3.5×10^{-4}
Ground Plane	2.1×10^{-5}	---------	---------	2.5×10^{-5}	--------
Total	2.6×10^{-4}	1.0×10^{-3}	4.1×10^{-5}	2.5×10^{-5}	3.5×10^{-4}
Nearest Milk Goat[c]					
Inhalation	1.6×10^{-7}	2.3×10^{-6}	5.5×10^{-7}	--------	6.5×10^{-7}
Goat Milk	1.2×10^{-4}	1.2×10^{-3}	1.5×10^{-4}	--------	1.4×10^{-3}
Ground Plane	1.4×10^{-5}	---------	---------	1.6×10^{-5}	---------
Total	1.4×10^{-4}	1.2×10^{-3}	1.5×10^{-4}	1.6×10^{-5}	1.4×10^{-3}
Nearest Milk Cow and Garden[b]					
Inhalation	7.1×10^{-7}	8.8×10^{-6}	1.2×10^{-6}	--------	1.5×10^{-6}
Vegetable	2.4×10^{-4}	1.0×10^{-3}	4.0×10^{-5}	--------	3.5×10^{-4}
Cow Milk	7.3×10^{-5}	4.2×10^{-4}	4.4×10^{-5}	--------	3.8×10^{-4}
Ground Plane	2.1×10^{-5}	---------	--------	2.5×10^{-5}	--------
Total	3.4×10^{-4}	1.5×10^{-3}	8.5×10^{-5}	2.5×10^{-5}	7.3×10^{-4}

(a) Maximum exposed individual is the adult.
(b) Maximum exposed individual is the child.
(c) Maximum exposed individual is the infant.

TABLE 2-4

DOSE TO INDIVIDUALS FROM ALL PATHWAYS
FOR TRITIUM RELEASES
(13 WEEK CONTINUOUS PURGE)

Location/Pathway	Dose (mrem)	
	Total Body	Lung
Nearest Residence[a]		
Inhalation	1.0×10^{-2}	1.0×10^{-2}
Ground Plane	------	------
Total	1.0×10^{-2}	1.0×10^{-2}
Nearest Garden[b]		
Inhalation	5.7×10^{-3}	5.7×10^{-3}
Vegetable	1.2×10^{-2}	1.2×10^{-2}
Ground Plane	-------	-------
Total	1.8×10^{-2}	1.8×10^{-2}
Nearest Milk Goat[c]		
Inhalation	1.9×10^{-3}	1.9×10^{-3}
Goat Milk	8.8×10^{-3}	8.8×10^{-3}
Ground Plane	-------	-------
Total	1.1×10^{-2}	1.1×10^{-2}
Nearest Milk Cow and Garden[b]		
Inhalation	5.7×10^{-3}	5.7×10^{-3}
Vegetable	1.2×10^{-2}	1.2×10^{-2}
Cow Milk	4.8×10^{-3}	4.8×10^{-3}
Ground Plane	------	------
Total	2.3×10^{-2}	2.3×10^{-2}

(a) Maximum exposed individual is the teen.
(b) Maximum exposed individual is the child.
(c) Maximum exposed individual is the infant.

TABLE 2-5

BONE DOSE TO CHILD AT
THE NEAREST GARDEN
USING FINAL PEIS METEORAOLOGICAL DATA

PATHWAY	BONE DOSE (mrem)
Inhalation	1.9×10^{-5}
Vegetable	4.4×10^{-4}
Ground Shine	---
Total	4.6×10^{-4}

3.0 OCCUPATIONAL EXPOSURE

3.1 External Exposures

All individuals entering the reactor building will be monitored for external exposures in accordance with GPU Radiological Control Procedures (RCP) to ensure personnel exposures are maintained within 10 CFR 20 dose equivalent quidelines. Administrative control points in accordance with GPU Procedures will be used in order to assure specified dose limits are not exceeded. Extremity monitoring will be performed in accordance with existing procedures.

The total exposure for the decontamination experiment is estimated to be 60 to 180 man-rem. This is based upon the scope as given in the decontamination experiment scope document referenced in Section 1 and includes area preparations, the decontamination experiments, post-experiment cleanup operations, experiment data acquisition tasks, health physics support, and the installation of a man-lift to provide access to the polar crane.

The man-rem estimate was calculated as follows. The estimated man-hours for each elevations for the above activities are 117 man-hours for elevation 305', 211 man-hours for elevation 347', and 80 man-hours for the polar crane. Based on past experience, estimated composite dose rates are 0.45 R/hour for elevation 305', 0.23 R/hour for elevation 347', and 0.15 R/hour for the polar crane. The resultant man-rem estimates are 53 man-rem for elevation 305'; 49 man-rem for elevation 347'; and 12 man-rem for the polar crane. This yields 114 man-rem for all activities. Because of the uncertainty in the dose rates and man-hours, the man-rem for the activities are estimated to vary by $\pm$ 50 percent. Considering the uncertainties associated with the man-rem estimate, 60 to 180 man-rem has been selected to be used as the estimate for the decontamination experiment.

Based on past experience, the whole body dose is expected to be the limiting exposure. Specifically, exposure data from several recent entries were reviewed and the following results were obtained. The average skin dose was 0.5 percent of the 10CFR20.101(a)(3) limit while the maximum exposure was only 3 percent of this limit. This is in contrast to the whole body doses where the average exposure was 25 percent of the 10CFR20.101(a)(1) limit (or 10 percent of the 3 rem/quarter limit) and the maximum exposure was 63 percent of the 10CFR20.101(a)(1) limit (or 26 percent of the 3 rem/quarter limit). These are based on TLD badges located on the chest where data for comparative purposes is most complete. The decontamination experiment is not expected to alter these results since protective clothing requirements and the radiological environment are not expected to be effectively different than previous entries. Consequently, the whole body dose has been and is expected to be the most limiting exposure. These data are consistent with the relatively low concentrations of Sr-90 compared to Cs-137 and the skin exposures to be expected from this source of exposure. The extremity exposures do not differ significantly from the whole body results. Consequently, they are not expected to be limiting either.

3.2 Internal Exposures

Personnel entering the reactor building will be protected against the inhalation of gaseous or particulate radioactivity in accordance with GPU Radiological Control Procedures.

As specified by Regulatory Guide 8.15, analyses of expected airborne contamination levels will be performed in order to select appropriate respiratory protective devices.

Air sampling for particulate activity will be performed using devices such as lapel samplers and methods such as grab samples. Tritium air samples will not be taken unless deemed necessary by bioassay, engineering judgement, or other substantive basis.

An estimate of the airborne radioactivity to be encountered by the individuals performing the decontamination experiment was derived from the BZA results of workers participating in the gross decontamination test of May 14, 1981. The average BZA concentrations of Cs-134 and Cs-137 are shown below by job function.

	Cs-134	Cs-137
Area preparation (4 workers)	2.9×10^{-8} µCi/cc	2.6×10^{-7} µCi/cc
Decontamination (3 workers)	2.0×10^{-9} µCi/cc	2.0×10^{-8} µCi/cc
Post-decon area clean-up (4 workers)	6.8×10^{-10} µCi/cc	6.6×10^{-9} µCi/cc

These results indicate that the decontamination test did reduce the airborne activity of Cs-134 and Cs-137. Similar results can be expected on the upcoming decontamination experiment. It is also expected that Sr-90 concentrations will exhibit trends comparable to Cs-134 and Cs-137.

Although airborne radioactivity is expected to decrease as a result of the decontamination experiment, present concentrations of these isotopes (8.6×10^{-9} µCi/cc of Cs-134, 9.1×10^{-8} of Cs-137 and 1.7×10^{-9} of Sr-90) would yield no problems in respiratory protection. Estimated MPC hours are 0.012/hour with air purifiers (PF 1000) using current concentrations.

Tritium levels are not expected to pose difficulties. Recent tritium levels in the Reactor Building have been maintained well below MPC levels by purging the Reactor Building. Recent bioassay results have indicated uptakes which would cause exposures of less than 5 mrem.

3.3 Measures Taken to Reduce Occupational Exposure to As Low As Is Reasonably Achievable (ALARA) Levels.

The objective of minimizing occupational exposure has been a major goal in the planning and preparation for the decontamination experiment. The actions that have been taken or are being planned toward meeting this objective are summarized in this section.

Protective clothing and respirators will be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

The decontamination experiment is designed to accomplish two distinct goals:

1. Provide, by experimental means, a substantial data base and operating feedback for the planning and execution of the subsequent decontamination activities in the TMI-2 containment.

2. Achieve significant decontamination of the floors and other horizontal surfaces on elevation 305' and 347', and the polar crane.

While all the techniques used to accomplish goal No. 2 above are designed to achieve substantial decontamination, it must be pointed out that the activity is experimental in nature, designed to create a data base, and as such, some techniques may be utilized which are not the absolute optimum technique for accomplishing decontamination. However, the primary purpose of the experiment is to create experimental data for future planning activities.

While the proposed activity is experimental in nature, the techniques and sequence of operation chosen have been developed to achieve the greatest decontamination at minimum man-hour and man-rem expenditure in the containment. Data to date indicates that a substantial fraction of the contamination in the TMI-2 containment is associated with loose debris (e.g., dirt, rust, boric acid crystals, etc.) on horizontal surfaces. The major dose reduction benefit can, therefore, be achieved by removing the debris and its associated contamination from horizontal surfaces. As concluded from the results of the large scale decontamination test in May of 1981, one effective technique for removing this debris is high flow rate, low presssure water flushes. All the low pressure techniques specified for the decontamination experiment are designed to determine the optimum flow rates and temperatures for gross debris removal. The sequence of operations begins with a low pressure flush at the 305' elevation for dose reduction purposes. This is ALARA because all activities in the TMI-2 containment are initiated and staged from the 305' elevation regardless of their ultimate location. In addition, great benefits in the area of contamination control in the air lock and anteroom will undoubtedly result from this first step.

One of the principles which is used to determine the sequencing of operations is that no decontamination step should be invalidated by a subsequent step. Thus the sequence is developed such that at least one decontamination task will be executed on a horizontal surface following any previous step which could deposit contamination on that surface. This principle allows minor resequencing in the overall schedule of events but will not cause any decontamination activity to be invalidated. Thus at least one decontamination pass over the 347' elevation will remain in the sequence of activities following polar crane decontamination and the high pressure pass across the 305' elevation will be the final decontamination activity contained in the experiment.

Execution of the individual tasks within the decontamination experiment are maintained ALARA by the execution of the ALARA checklist and very substantial mockup training of work crews. This training will approximate the actual work situation as closely as can be achieved for each task utilizing appropriate equipment, protective clothing, and respiratory protection.

Extensive planning of tasks to be conducted in a radiation field, and training of personnel will be used to reduce the time needed to complete a task. Extensive use of photographs and the in-containment closed circuit television system will be used to familiarize personnel with the work area. The higher radiation areas are identified to personnel and the work is structured to avoid these areas to the extent practical. Practice sessions will be utilized as necessary to ensure that personnel understand their assignments prior to entering the containment. Planning and training are proven methods of ensuring that personnel are properly prepared to conduct the assigned task expeditiously.

The occupational exposure received will also be minimized by judicious use of temporary services or minor modifications to existing systems in the containment rather than installing a new system in the containment to accomplish the same function.

The total duration of decontamination activities in the decontamination experiment is estimated to be about two weeks. Experience to date indicates that once the decontamination activity is initiated it can be most expeditiously concluded by a continuous progressive work activity with minimum interruption or perturbation of the task. It is believed that ALARA benefits which might be achieved from continual minor course corrections would be more than offset by the inefficiencies introduced by continual retraining and further segmenting of the work activity.

As the decontamination experiment progresses, techniques which are obviously grossly unsuccessful, or which produce effects outside certain ground rules (e.g., substantial damage to coated surfaces from high pressure decontamination techniques) will be abandoned or modified prior to proceeding. Techniques which display acceptable levels of success and whose results remain inside the operational boundaries which have been established for the experiment, will be pursued to completion as scheduled. Information and data will be acquired during the course of the experiment and evaluated as expeditiously as possible to provide feedback on which experiment modifications may be based. Any substantive feedback will have to be rapid and definitive if it is to be used to alter the proposed course of action during the experiment. Three techniques will be used to collect information and data.

1. Closed circuit television observations of the decontamination activities.

2. Detailed post entry debriefings of the work crews.

3. General area radiation surveys and smearable contamination measurements. (It should be noted that the results of smearable contamination measurement early in the experiment will give qualitative information at best. The amount of debris on the horizontal surfaces of the TMI-2 containment, and the

levels of contamination associated with this debris make loose contamination smears extremely inefficient and very difficult to interpret.)

The information gained by the observation techniques described above will be reviewed on an entry by entry basis by decontamination and data acquisition supervisory personnel. Potential improvements in operational technique will be fed back into future work packages and mockup training in a manner consistent with the development of work activities. If the observation techniques definitively demonstrate major operational problems, or the ineffectiveness of a particular decontamination technique, the decontamination experiment schedule and content can be altered to properly accommodate this feedback. It should be noted, however, that the evaluation of the adequacy of a particular decontamination technique must take into account and weigh several operational factors such as man-rem and man-hour expenditure, personnel safety, operational complexities and training requirements, etc. As a result of this weighted evaluation, the most effective decontamination technique may not be the most efficient technique on the basis of decontamination effectiveness per unit effort or expenditure. Again, it must be emphasized that decontamination experiment is designed to provide the data base from which decisions may be made later regarding the most efficient and optimized decontamination sequences and techniques.

4.0 RADIOACTIVE WASTE MANAGEMENT

4.1 Solid Waste

An estimate of the solid waste material which will be generated as a result of the reactor building decontamination experiment has been made. Seperate waste categories were established as follows:

1. Disposable Protective Clothing - includes gloves, shoe covers and wet suits which will be utilized by personnel preparing the reactor building and actually conducting the experiment.

2. Reactor Building Trash - this category consists of the accumulated trash in the reactor building which must be removed prior to the start of the experiment.

3. Submerged Demineralizer System and EPICOR-II - consists of the number of liners of ion exchange material which will be generated by processing the flush water used in the experiment.

4. Miscellaneous Waste - includes plastic bags, strippable coating, framing lumber, polyethylene sheet and other disposable equipment used in support of the decontamination experiment.

Table 4-1 give the estimated quantities for each category of solid waste. This estimate does not include waste from the laundering of reusable protective clothing, wastes resulting from the decontamination of tools or equipment, or equipment which will not be decontaminated, but will be retained for reuse.

Solid waste will be disposed of in accordance with established procedures.

4.2 Liquid Waste

An 80,000 gallon supply of processed water is available for the decontamination experiment. Most of the water actually used for the experiment will drain through the flow drains and be collected in the containment sump area. This water will be processed through the submerged demineralizer system with the water presently in the sump.

TABLE 4-1

ESTIMATED QUANTITIES OF

SOLID WASTE AS A RESULT OF THE

DECONTAMINATION EXPERIMENT

Waste Form	Quantity
Disposable Protective Clothing	110 cubic feet (uncompacted)
Reactor Building Trash	110 cubic feet
Submerged Demineralizer System and EPICOR II	less than one liner from each system
Miscellaneous Waste	100 cubic feet

5.0 INDUSTRIAL SAFETY

5.1 Fire Protection

In order to reduce the likelihood of a fire in the Reactor Building during the decontamination experiment, the following precautions will be implemented:

a. Transient combustible material will be kept to a minimum in the Reactor Building.

b. All activities which increase the likelihood of a fire such as welding, burning or grinding will be reviewed and controlled in accordance with plant procedures.

Operability of the fire detection system installed in the containment is questionable. Therefore, fire detection will be via closed circuit television monitoring during all activities and via personal observation by fire watches posted during the activities listed in b, above. Fire watches will have radio communication with the command center and will be equipped with the appropriate portable fire extinguisher.

All personnel are equipped with small flashlights, for emergency lighting and both airlocks are available for ingress and egress with No. 2 Airlock being the normal path. The GAI-tronics page system in the containment is operational and can be used to notify entry personnel.

5.2 Personnel Protection From High Pressure Water Spray

High pressure water sprays have been widely used in the nuclear and chemical industries for surface and equipment cleaning. High pressure water sprays of about 1000 to 2000 psi were demonstrated in the containment decontamination test. The results of this test indicate that a higher water pressure, higher flow rate spray can be more effective for some operations than a low pressure, low flow rate spray. The maximum expected water discharge pressure to be used for the decontamination experiment is 6000 psi. The maximum capability of the high pressure water spray pump is 10000 psi at 25 gpm. This equipment was selected for its flexibility and for use in future tests.

Personnel will receive extensive training and instruction in the proper use of high pressure sprays to prevent personnel injury. In addition, the equipment is designed with features which minimize the potential for operator injury. Personnel will also be provided with protective equipment.

6.0 SAFETY EVALUATION FOR CONTAINMENT DECONTAMINATION EXPERIMENT

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

None of the activities associated with the decontamination experiment will affect the condition of the reactor coolant system or the fuel. The core is being maintained in a subcritical condition by the boron concentration in the reactor coolant. None of the activities that will occur during the decontamination experiment will affect the boron concentration. The safety-related equipment required for the loss-to-ambient cooling mode of decay heat removal will not be altered during the experiment.

The decontamination experiment will not increase the probability of occurrence or the consequences of an accident previously evaluated in the FSAR.

The decontamination experiment does not create the possibility for an accident different than any evaluated previously in the FSAR.

The decontamination experiment will not require a technical specification change.

The decontamination experiment will not reduce the margin of safety as described in the bases for any technical specification.

Therefore, the decontamination experiment does not involve an unreviewed safety question as defined in 10 CFR Part 50, paragraph 50.59.

a) nearest residence (0.5 mile, E)

b) nearest garden (1.1 mile, ENE)

c) nearest milk goat (1.2 mile, N)

d) nearest milk cow and garden (1.1 mile, E)

7.0 CONCLUSION

Based upon the Radiological and Safety Evaluations contained in this report, is concluded that:

1) Offsite releases and doses for the decontamination experiment are well within the bounds of the TMI-2 Technical Specification limits, even making very conservative assumptions and incorporating a larger scope of activities than the decontamination experiment.

2) Occupational exposures to preform the decontamination experiment are consistent with ALARA considerations, and

3) The decontamination experiment does not constitute an unreviewed safety question as defined by 10 CFR 50.59.