RADIOLOGICAL AND SAFETY

EVALUATION OF ONGOING CONTAINMENT BUILDING

DECONTAMINATION ACTIVITIES

FOR

TMI-2 RECOVERY

REVISION 0

1.0 SCOPE

The information in this document represents the radiological and safety evaluation of decontamination activities to be performed in the reactor containment for elevation 305 and up. Included in Section 2.0 is a description of planned decontamination activities, which include the current assessment of the most appropriate decontamination methods. These methods may change, based on the results from future decontamination operations. If these methods change substantially from those presented here, an engineering evaluation will be performed to determine the effects of the change on the areas presented within. This document addresses the Ongoing Containment Decontamination activities for the 52 week period following the approval of this document.

Sections 3.0 thru 7.0 present an evaluation of the radiological and safety aspects of the planned operation.

Included in the evaluation are the following:

a) effluents to the environment,
b) occupational exposures,
c) radioactive waste management,
d) industrial safety, and
e) safety evaluation (10 CFR 50.59).

The evaluation concludes that the proposed activities can be accomplished with minimal impact on the health and safety of the public.

2.0 ONGOING CONTAINMENT DECONTAMINATION ACTIVITIES

2.1 General

In March of 1982 a decontamination experiment was performed which resulted in removal of sizeable quantities of loose contamination from the 305'-0" and 347'-6" elevations of the Reactor Building. The primary techniques identified for use in containment were low pressure water flushing and high pressure spraying (excluding the 282'-6" elevation). In order to determine added decontamination effectiveness, floor scrubbing and wet vacuuming were tested at the conclusion of the decontamination experiment.

The results of the decontamination experiment showed that the low pressure water flushing did decrease airborne contamination and removed the largest visual particulate deposits. The high pressure water spray removed additional contamination (both loose and within the surface film), but the combination of the above two techniques could not, on the average, reduce the smearable contamination to less than 10^5-10^6 dpm/100cm^2.

Tests with the floor scrubber, used in combination with a wet-vacuum, did show additional reduction in levels of contamination. For the test situation, smearable contamination was reduced to the range of 10^3-10^4 dpm/100cm^2.

The approach to remove additional contamination from surfaces on the 305'-0" and 347'-6" elevation of the Reactor Building is based on the results of the decontamination experiment and an engineering evaluation. Essentially the approach is to flush a surface with water to remove gross levels of contamination and then follow up with a secondary technique such as floor scrubbers, abraisive pads, or wet vacuuming to further reduce levels. Water levels in the sump will be controlled such that the reserve tankage limits specified in the Operating License are not exceeded. Strippable coatings will then be applied where appropriate to fix remaining surface contamination and aid in contamination control. Prior to using any chemicals except as discussed below, an engineering evaluation will be performed and provided to the staff to ensure that there will be no adverse impacts to equipment or to the health and safety of the public. Although there is no large scale use of chemicals presently planned, any chemicals used will be evaluated as discussed above and placed on an approved list prior to usage. Chemicals will be controlled using a wet vacuum, wipes, or other approved method.

2.2 Technique Selection

According to the results of the Decontamination experiment and engineering evaluation the sequence of operations for reactor building decontamination should minimize contamination deposited on lower elevations as a result of liquid drippage or solid particulates settling out from decontamination of higher surfaces. According to this approach, the following is presented in the suggested sequence along with a brief description of the decontamination approach. It should be realized that the sequence and the techniques presented represent the optimum based upon current knowledge. If alternate sequences or techniques are determined to be more expeditious or exposure conservative they will be used. It is not expected that such alternatives will result in significant changes from the information presented in this document.

- Flush the Reactor Building Dome.

  The intent of this operation is to perform a low velocity, high volume flush of the dome surface. Due to equipment limitations, high volume is limited to 25 gpm and high velocity is limited by nozzle design and the ability to maintain a uniform flush pattern at a distance of 30 to 60 feet from the nozzle. The goal for dome decontamination is to reduce loose materials that may be dislodged from the dome surface during later defueling activities requiring operation of the polar crane.

- Flush the Polar Crane.

  The intent of this operation is to perform a low velocity, high volume flush of the platform and other major surfaces of the polar crane. The same low pressure flush parameters and equipment as was used for dome flushing will be applied. The goal for polar crane decontamination by flushing is to remove loose particulates (and oils, if possible) from horizontal surfaces to reduce personnel contamination potential during crane refurbishment. Prior to flushing the crane, free standing oils and grease will be removed by wiping where physically possible. In addition, the flushing will be required to remove collected debris that may have been loosened from the Reactor Building dome during flushing.

- Scrub Selected Components of Polar Crane.

  Due to crane refurbishment requirements, specific components will be made accessable for repair or replacement. Prior to extensive personnel handling, a decontamination which would remove the majority of loose particulates and oils would reduce personnel contamination potential. Since scrubbing with a pad worked well on the 347'-6" elevation floor during a test conducted as part of the decontamination experiment, a similar manual scrubbing will be performed. The scrub pad will either be cloth or abrasive pads used with either demineralized water or an approved chemical solution to emulsify the oil. Following scrubbing, cloth wipe will be used to collect any chemicals and/or remaining particulates.

- Flush Vertical Surface and Equipment on 347'-6" Elevation.

  A flushing operation will be performed to remove particulate debris that may have come down during flushing of the dome and polar crane. This flushing will be the same as that performed during the decontamination experiment.

- Flush and Scrub Floor on 347'-6" Elevation.

  An initial flushing operation will be performed to remove particulate debris that may have come down during flushing of the dome and polar crane. Following flushing a mechanical floor scrubber will be used on the floor surface to remove additional contamination.

- Decon enclosed Stairwell and Elevator Shaft Down to the 305'-0" Level.

  Since removal of contamination in the upper part of the building is needed for contamination control, the elevator shaft and stairwell No. 2 should be flushed.

The flushing operation is by application of 25 gpm processed water with a spray device that should effectively rinse most surfaces within stairwell No. 2 and the elevator shaft. The flushing device will be used throughout the stairwell and elevator shaft down to the 305'-0" elevation.

- Flush Overheads on 305'-0" Elevation.

  During the decontamination experiment, the overheads were flushed. This was done manually with a wand at the 305'-0" floor level. Since this flushing removed considerable dirt and debris a more thorough flushing is required to remove remaining materials. A flushing will therefore be performed with a 25 gpm spray device located at an elevation within the overheads. This spray device will be used remotely by mounting of the device, then having personnel leave the Reactor Building during spraying.

- Flush verticals and equipment on 305'-0" Elevation.

  Flushing of the overheads in paragraph above will result in recontamination of the verticals and equipment on the 305'-0" elevation. These recontaminated surfaces will require flushing to remove this contamination.

- Flush and Scrub Floor on 305'-0" Elevation.

  An initial flushing operation will be performed to remove particulate debris that may have contaminated the floor during flushing of the overheads. Following flushing, a mechanical floor scrubber will be used on the floor surface to remove additional contamination. When used in conjunction with the wet-vacuum, this technique proved to effectively reduce smearable contamination during the decontamination experiment performed on the 347'-6" elevation. The mechanical scrubber will either utilize a very abrasive scrub pad with demineralized water or an approved chemical solution.

- Decon Service Structure.

  Previous radiation and swipe measurements have indicated considerable levels of contamination exist within the reactor head service structure. A 25 gpm low pressure processed water flush should remove the majority of the loose contamination. This contamination is expected to be similar to the type of contamination that was effectively removed from refueling canal surfaces by water flushing during the decontamination experiment.

- Flush LOCA Ducts.

  In order to remove recontamination potential from the polar crane area, the LOCA duct internals are to be flushed with low pressure water. This flushing will involve the use of a spraying mechanism that will direct water flow at 25 gpm to the internal surfaces.

- Flush D-ring Interior.

  Access to the systems within the D-rings is required for preparations for reactor inspections. In addition airflow from the aircoolers is directed up the D-rings and a flushing operation should remove loose particulates that may contribute to the recontamination potential. The flushing operation is by application of 25 gpm processed water with a spray device that should effectively rinse most surfaces within the D-ring. The flushing device will be used at several locations to accomplish this.

2.3 Decon Effectiveness Measurements

Effectiveness of the decontamination operations will continuously be monitored frequently to evaluate the progress and determine if changes are required. The majority of the measurements will be in accordance with GPUN Radiological Controls procedures. Additional measurements may be required as identified by the specific work task description.

Exposures for the major Ongoing Containment Decontamination activities (i.e., decon polar crane, service structure, etc.) shall be tracked against established exposure targets with the Exposure Management Program and will be monitored and reviewed by Radiological Engineering personnel.

2.4 Contamination Control

In order to prevent recontamination of the surfaces decontaminated with the abrasive scrubber, a strippable coating will be applied. In addition, selected wall and equipment surfaces will be coated to aid in reduction of airborne contamination.

Following completion of the decontamination program identified above, ongoing decontamination maintenance may be required. The methods previously discussed will be used on an as-needed basis.

3.0 OFFSITE RADIOLOGICAL DOSES

3.1 Effluents to the Environment

During the performance of the activities 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 containment 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 Section 3.1.1 and 3.1.2 respectively.

The assumptions used in the calculations of effluents are:

1) The purge will be in continuous operation for 365 days.

2) the containment purge rate is assumed to be 25,000 CFM. Even if the purge were to be operated for some limited period of time at 50,000 CFM, it is expected that the additional releases would still be within the bounds of this analysis.

3.1.1 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-219A

2) Grab Samples

3) BZA Data

Measurements of airborne activity in the containment atmosphere during the Decontamination Experiment were reviewed and mean particulate airborne concentration of each isotope was determined. The concentration for the various isotopes are given in Table 3-1.

It is assumed the particulate airborne concentration as given in Table 3-1 remain constant throughout any containment entry. This means it is assumed that there is no change in airborne particulate activity as a result of activities being performed in the containment. This is a conservative assumption since each subsequent decontamination activity will reduce overall activity to some degree. The results of the decontamination experiment conducted in March, 1982, show a marked reduction in airborne particulate activity. Also, it is assumed the containment purge is operated continuously. An additional conservative assumption is that the concentrations obtained during the Decontamination Experiment from local breathing zone air samplers are assumed to be uniformly distributed throughout the entire containment volume.

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 3-2.

Using the results given in Table 3-2, the resulting doses to individuals were calculated in accordance with the guidance provided in Regulatory Guide 1.109. The calculated doses are given in Table 3-3.

This analysis uses the meteorological data (X/Q and D/Q) presented in the Offsite Dose Calculation Manual (ODCM).

3.1.2 Tritium Release

Measurements of the tritium concentration in the containment atmosphere during the Decontamination Experiment from grab samples in the work areas were reviewed and an average airborne concentration was evaluated to be 2.5E-6 μCi/cc for periods of incontainment activity. Tritium concentrations for ambient air is derived from the numerical average from the last 20 entries. This value is 1.0E-6 μCi/cc. Using these values for the source terms, the quantity of tritium released to the environment was calculated based on the following assumptions:

a) A concentration of 2.5E-6 μCi/ml was assumed to exist for 1200 hrs. and a concentration of 1.0 E-6 μCi/ml was assumed for the rest of the year (7560 hrs.).

b) The containment purge exhaust is operated continuously at 25,000 CFM for the year.

The quantity of tritium released was calculated to be 450 Ci. Using the value of 450 Ci and the guidance provided in Regulatory Guide 1.109, the resulting doses to individuals were calculated. These results are presented in Table 3-3.

3.1.3 Discussion of Results

Particulate Releases

If the offsite doses given in Table 3-3 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 which include all sources, it can be concluded these calculations are enveloping for decontamination activities. 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. Actual decontamination activities which occurred during the Decontamination Experiment in March of 1982, resulted in values measured at HPR-219A that were not greater than the below listed lower limits of detection.

Cs-134	<2.0E-14 μCi/cc
Cs-137	<2.0E-14 μCi/cc
Sr-90	<2.0E-14 μCi/cc

This means that when activities were being conducted in the containment similar to those to be conducted during the ongoing containment decontamination, no detectable releases were measured which could be directly attributed to those activities. Therefore, it is reasonable to expect the decontamination activities will not result in any measurable increase in releases or offsite doses.

Tritium Releases

The calculatd release for tritium from the described activities is 450 Ci. This results in a calculated offsite dose of 1.78E-2 millirem (see Table 3-3). This dose is small when compared to the limits given in Appendix B of the Technical Specifications.

TABLE 3-1

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

Radionuclide	Concentration (μCi/cc) $\pm$ 1(SD)
Cs-134	4.6E-9 $\pm$ 2.1 E-9
Cs-137	5.4E-8 $\pm$ 2.6 E-8
Sr-90	4.8E-9 $\pm$ 5.0 E-9

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.

Typically, an air sample taken at 4 CFM for 15 minutes will normally have associated the following LLD's for a 1000 second count:

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

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 2.5E-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 3-2

CALCULATED PARTICULATE AIRBORNE RELEASES TO THE ENVIRONMENT

Radionuclide	Release (Ci)
Cs-134	1.71E-03
Cs-137	2.01E-02
Sr-90	1.78E-03

TABLE 3-3

DOSE TO MAXIMUM EXPOSED INDIVIDUALS FROM
ALL PATHWAYS FOR AIRBORNE RELEASES
(52 WEEK CONTINUOUS PURGE, 25,000 CFM)
ODCM METHODOLOGY

Nuclide	μCi/cc	Ci/yr	μCi/sec	mRem/yr
Cs-134	2.3E-9	1.71E-3	5.4E-4	2.21E-1
Cs-137	2.7E-8	2.01E-2	6.35E-4	2.34E-0
Sr-90	2.4E-9	1.78E-3	5.65E-5	3.75E-1
H-3	*	4.56E2	1.43E1	1.78E-2

* μCi/cc vary based on the assumptions discussed on Page 7 (i.e. 2.5E-6 μCi/cc, for 1200 hours, and 1.0E-6 μCi/cc for the remainder of the year (7560 hrs.).

4.0 OCCUPATIONAL EXPOSURE

4.1 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 limits. Administrative dose limits in accordance with GPU Procedures will be used in order to assure that 10 CFR 20 dose limits are not exceeded. Extremity monitoring will be performed as needed in accordance with existing procedures.

The assumptions used in the calculation of occupational exposures are:

1) The in-containment man-hours to support the ongoing containment decontamination are 1,800.

4) The in-containment radiation dose rates and airborne activity levels remain constant throughout in-containment decontamination activities.

The total exposure for the ongoing containment decontamination activities is estimated to be 180 to 550 man-rem. This is based upon general decontamination activities and includes area preparation, decontamination activities, cleanup operations, periodic sampling, health physics support, installation of necessary equipment and any other activity necessary to support decontamination operations.

The man-rem estimate was calculated as follows. Based on past experience, estimated composite dose rates are 0.35 R/hour (720 hrs.) for (540 hrs.) elevation 305', 0.12 R/hour (540 hrs.) for elevation 347', and 0.10 R/hour for the polar crane and dome. The resultant man-rem estimates are 252 man-rem for elevation 305'; 65 man-rem for elevation 347'; and 54 man-rem for the polar crane and dome. This yields 371 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, 180 to 550 man-rem has been selected to be used as the estimate for the next year of the Ongoing Containment Decontamination program.

Table 4.1 summarizes the estimated occupational exposure for the next year of the Ongoing Containment Decontamination program. A review of entry dosimetry results indicate that the deep dose equivalence will be the limiting exposure, with the dosimeter located on the thigh 6 inches above the knee recording the largest deep dose equivalence value.

4.2 Internal Exposures

Personnel entering the Reactor Building will be protected against the inhalation of gaseous or particulate radioactivity as necessary 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 be taken unless deemed unnecessary by the GPU Radiation Controls Department by bioassay, engineering judgement, or other substantive basis.

An estimate of the airborne radioactivity to be encountered by the individuals performing decontamination activities was derived from the BZA results of workers participating in the Decontamination Experiment performed in the first quarter of 1982. The average BZA concentrations of Cs-134, Cs-137 and Sr-90 are shown below by job function.

	Cs-134	Cs-137	Sr-90
Area Preparation	3.0E-8 μCi/cc	1.9E-7 μCi/cc	3.0E-9 μCi/cc
Decontamination	4.6E-9 μCi/cc	5.4E-8 μCi/cc	4.8E-9 μCi/cc
Post-decon Activities	3.0E-9 μCi/cc	4.7E-8 μCi/cc	1.4E-9 μCi/cc

These results indicate that the decontamination did reduce the airborne activity of Cs-134, Cs-137 and Sr-90. Similar results can be expected on the upcoming decontamination activities.

Although airborne radioactivity will decrease as a result of the decontamination activities, the concentrations of these isotopes (4.6E-09 μCi/cc of Cs-134, 5.4E-08 of Cs-137 and 4.8E-09 μCi/cc of Sr-90) during the decontamination activities will yield no problems in respiratory protection. Estimated MPC-hours are 0.01 MPC-hours/hour with air purifiers (PF 1000) using the above conservative concentrations. Tritium levels are not expected to pose difficulties. Recent bioassay results from persons participating in the Decontamination Experiment have indicated uptakes which would result from exposures to a mean tritium airborne activity level of 1.7 E-06 μCi/cc or 0.34 mpc-h per hour.

4.3 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 all activities in the containment. 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.

Decontamination activities are designed to accomplish goals:

1. Reduce loose surface beta-gamma contamination levels on floors to less then 5 X 10E3 dpm/100cm^2.

2. Reduce loose surface beta-gamma contamination levels on overheads to less then 1 X 10E5 - 1 X 10E6 dpm/100cm^2.

The techniques and sequence of operations chosen have been developed to achieve the greatest decontamination at minimum man-hour and man-rem expenditure in the containment.

Execution of individual decontamination tasks are maintained ALARA by a detailed radiological review by Radiological Engineering 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.

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 activities shall 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.

TABLE 4.1

OCCUPATIONAL EXPOSURE ESTIMATES
ONGOING CONTAINMENT DECONTAMINATION

	MEAN DOSE	MAX DOSE	COLLECTIVE DOSE
Deep Dose Equivalence	2.0 Rem	4.0 Rem	371 Rem
Shallow Dose Equivalence	2.0 Rem	4.0 Rem	371 Rem
MPC-hours (Particulate)	0.1	0.2	18.0
Dose Equivalence (Tritium)	0.009 Rem	0.017 Rem	1.53 Rem

5.0 RADIOACTIVE WASTE MANAGEMENT

5.1 Solid Waste

An estimate of the solid waste material which will be generated as a result of the Reactor Building decontamination activities has been made based on experience gained in previous Reactor Building entries. Separate 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 decontamination.

2. Reactor Building Trash - this category consists of the accumulated trash (e.g. plastic bags, framing lumber, polyethelene sheets and other disposable equipment) in the Reactor Building which must be removed prior to decontamination of the staging areas. Sources for this trash include initial construction materials and Recovery Construction activities.

3. Submerged Demineralizer System and EPICOR II - consists of the volume of liners of ion exchange material and filters which will be generated by processing the flush water used for decontamination.

4. Miscellaneous Waste - includes material for hand wiping surfaces, plastic bags, strippable coating, framing lumber, polyethylene sheet and other disposable equipment used in support of the decontamination activities.

Table 5-1 gives the estimated quantities and curie content for each category of solid waste. Curie estimates are based on experience from past Reactor Building entries. 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 classified and disposed of in accordance with established procedures.

5.2 Liquid Waste

A maximum of 200,000 gallons of processed water will be used for the decontamination. When operational flexibility permits, processed water with the lowest concentrations of radionuclide will be used. Most of the water actually used for decontamination 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 5-1

ESTIMATED QUANTITIES OF
SOLID WASTE AS A RESULT OF THE
ONGOING CONTAINMENT DECONTAMINATION

Waste Form	Quantity	Curie Content
Disposable Protective Clothing	336 cubic feet	23 Ci
Reactor Building Trash	440 cubic feet	30 Ci
Submerged Demineralizer System and EPICOR II	2149 Cubic Feet*	**
Miscellaneous Waste	400 cubic feet	42 Ci

*Based upon 200,000 gallons of water used and includes the volume of the liners.

**System efficiencies for the Submerged Demineralizer System and EPICOR II System are as discussed in the TER for the Submerged Demineralizer System.

6.0 INDUSTRIAL SAFETY

6.1 Fire Protection

In order to reduce the likelihood of a fire in the Reactor Building during the decontamination, 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.

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.

6.2 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 6000 psi were demonstrated in the containment decontamination experiment. The results 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 is 6000 psi. The maximum capability of the high pressure water spray pump is 10,000 psi at 25 gpm.

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.

6.3 Use of Ice Vests In Containment

In order to reduce the experienced problem with heat stress the following criteria shall be used in determining the mandatory use of ice vests by personnel entering the containment. This applies to all personnel wearing full impermeable clothing with or without backs cut out of the tops.

Containment access personnel shall contact the Control Room on the entry day for Reactor Building temperatures and specify ice vest requirements in accordance with Table 6-1.

TABLE 6-1

Criteria for Use of Ice Vests in

Unit II Containment Building

Reactor Building Temperature	Anticipated Stay Time	Ice Vest Requirements
Less than 70°F	1 hour or less greater than 1 hour	None (optional) 1/2 loaded vest
70° - 80°F	1 hour or less greater than 1 hour	None (optional) full loaded vest
80° - 90°F	45 minutes or less greater than 45 minutes	None (optional) full loaded vest
90° and above	All stay times	full loaded vest

The Safety and Health staff continuously monitors this program, modifications and improvements are made on a continuing basis. Other safety equipment may be utilized should evaluations show an improvement in personnel protection.

7.0 SAFETY EVALUATION FOR THE ONGOING CONTAINMENT DECONTAMINATION ACTIVITIES

Changes, Test, 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 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 Ongoing Containment Decontamination.

None of the activities associated with the Ongoing Containment Decontamination 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 will affect the boron concentration in the reactor. Boron concentrations in the water used in the decontamination will be maintained at or above 1700 ppm per NRC approved procedure. The safety-related equipment required for the loss-to-ambient cooling mode of decay heat removal will not be altered during the decontamination.

The decontamination will not increase the probability of occurrence or the consequences of an accident previously evaluated in the FSAR and/or other Safety evaluation submitted on the docket. The decontamination does not create the possibility for an accident different than any evaluated previously in the FSAR and/or other Safety evaluation submitted on the docket. The decontamination will not require a technical specification change.

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

Therefore, the Ongoing Containment Decontamination activity does not involve an unreviewed safety question as defined in 10 CFR Part 50, paragraph 50.59.

8.0 CONCLUSION

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

1. offsite releases and doses for the Ongoing Containment Decontamination activity 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 previously performed decontamination experiment,

2. occupational exposures to perform the decontamination activity are consistent with ALARA considerations, and

3. the decontamination activities do not constitute an unreviewed safety question as defined by 10 CFR 50.59.