Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

September 29, 1983
4410-83-L-0227

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US 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 Evaluation Report for the Ongoing Containment Building
Decontamination and Dose Reduction Activities - September 1983

Attached for your review and approval is the Safety Evaluation for the Ongoing Containment Building Decontamination and Dose Reduction Activities. This Safety Evaluation updates and, when approved, supplants the "Radiological and Safety Evaluation of Ongoing Containment Building Decontamination Activities for TMI-2 Recovery" submitted on September 23, 1982. This updated Safety Evaluation addresses In-containment Decontamination and Dose Reduction Activities for the remainder of 1983 and all of 1984. Should a requirement for in-containment decontamination arise prior to approval of the attached evaluation, we request your concurrence to proceed under the prior approved Safety Evaluation.

Sincerely,

B K Kanga

B. K. Kanga
Director, TMI-2

BKK:JJB:TLG:djb

Attachment

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

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SAFETY EVALUATION REPORT

FOR THE

ONGOING CONTAINMENT BUILDING

DECONTAMINATION AND DOSE

REDUCTION ACTIVITIES

SEPTEMBER 1983

THREE MILE ISLAND

UNIT 2

Revision 0
September 21, 1983

1.0 SCOPE

The information in this document represents the radiological and safety evaluation of decontamination and dose reduction activities to be performed in the reactor containment for elevation 305 and up. Included in Section 2.0 is a description of planned decontamination and dose reduction 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 area presented within. This document addresses the Ongoing Containment Decontamination and Dose Reduction activities for the the remainder of 1983, and all of 1984 (Ref. 1).

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. Additional tests using dilute phosphoric acid with a floor scrubber on reactor building elevation 347' also showed reduction of smearable contamination to the same range. Other chemical trials on the canal seal plate also showed similar reductions.

The approach to remove additional contamination from surfaces of the Reactor Building is based upon the results of engineering evaluation and experience in the auxiliary building at the 280' level and the decontamination experiment. 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, abrasive 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.

Further progress in the overall containment decontamination program will require the use of more aggressive techniques. Therefore, new approaches will have to be tried to further reduce surface contamination levels. Among the approaches being considered are steam/vacuum cleaning techniques, chemical decontamination agents, and scabblers.

Use of agents such as phosphoric acid foam, sulfamic acid gel, nitric or citric acid complexes, and abrasive zeolites may prove effective for external surface area "facility" decon, internal piping "systems" decon, and interior equipment and/or "tank" decon. Specific chemicals are evaluated and controlled in accordance with GPUN Administrative Procedures and in accordance with Federal Regulations.

The potential large scale use of chemicals depends on engineering evaluations considered in conjunction with scheduled chemical demonstrations, design of processing equipment, and waste disposal. The results from demonstrations will be used to recommend appropriate chemicals based on decontamination effectiveness, waste stream manageability, and material compatibility. Additionally, chemicals will be chosen to ensure no adverse impact to occupationally exposed personnel and the health and safety of the public.

Although a complete systems decontamination of Reactor Building systems is in the future, some activity is anticipated soon. These operations would be performed to accomplish significant dose rate reduction or to operate or flush a system to support a scheduled recovery milestone.

In some areas of the Reactor Building, hot spot sources may now be identified as coming from piping. Those hot spots occurring in areas of significant occupancy will be evaluated for dose rate reductions. Tasks which accomplish this function include shielding, piping removal, filling and draining tanks, operational flush or flush with temporary equipment. In descending priority, the areas to be screened for dose rate reduction are:

- El. 347' outside D-rings and Reactor Head Area
- El. 305' outside D-rings
- Area Inside D-ring
- Area below El. 305'

Flushing equipment will be designed and operated consistant with ALARA principles, Nuclear Safety, and Plant Operating procedures. Descriptions of this equipment will be given in future Technical Evaluation Reports.

During the past year, there has been a significant recontamination problem in the containment. The recontamination is largely due to the airborne transport of contamination from highly contaminated surfaces. In addition, the use of borated water for flushing building surfaces has left a friable, powdery film of boric acid crystals which has been associated with the airborne activity concentrations measured on personnel samplers. This activity is caused by resuspension due to personnel movement. Three steps which are being planned for this period should have a positive effect on this situation. First will be the use of deborated water for flushes. Second will be the installation of chillers on the air coolers. While this is not a dose reduction activity, it should have the net effect of reducing this airborne contamination by 10% to 20%. The third planned step is to close the three dampers in ventilation ducts to the D-rings, to limit air circulation. Other steps that are being considered include decontamination of air vent duct work and isolating the levels from each other by sealing the floors through the seismic gap and other penetrations.

2.2 Planned Decontamination and Dose Reduction Activities

The following plans represent current thinking on Reactor Building Decontamination and Dose Reduction. These operations have been planned with ALARA considerations for occupational and off-site dose rates, and are based on previous experience with the decontamination experiment, previous experience with decontamination of the Reactor Building and the Auxiliary Building, and engineering evaluations. It should be realized that these operations represent the optimum based upon current knowledge. If alternate operations or techniques are determined to be more expiditious 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.

- Reflushing Reactor Building Surfaces with Deborated Water.

  All flushes done up to this time have used borated water which had been processed by the SDS/EPICOR II system. This led to the deposition of a thin layer of boron crystals on building surfaces, creating contamination and safety problems. These problems will be lessened by flushing these surfaces with processed, deborated water. Surfaces to be reflushed include the reactor building dome, the polar crane, the vertical surfaces and equipment on the 347'6" level, the floor on the 347'6" elevation, the overheads, verticals, equipment and floors on the 305' elevation, the LOCA ducts, and the D-ring exteriors. In addition to removing the layer of boron crystals, this should work to removing accumulated recontamination from those surfaces. Since these operations will be done after the planned reactor head lift, a covering will be placed over the reactor to prevent the deborated water from leaking into the open vessel. This step should be sufficient to keep the reactor coolant boron concentration from being diluted below the 3500 ppm required by the Technical Specifications.

- Rescrub 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.

- Hands On Decontamination of Vertical Surfaces on 347'6" Elevation.

  This is the follow-on to the flushing that was done last year. As well as being part of the general decontamination effort, this will also clean up any debris that may come down from the polar crane.

- Decontaminating of the Air Coolers

  The air cooler ductwork has already been flushed with borated water. The next step in the decontamination effort is to spray the inner surfaces with an acid foam, which would be removed by water flushing. The chemical waste would be contained in a positive isolation and taken out of the reactor building for separate processing. Presently, two disposal options are being considered: 1) anion exchange pretreatment for ultimate processing through the SDS/EPICOR system; or 2) direct solidification. After the acid-foam decontamination, the dose rate from the air coolers will be evaluated, and the air coolers will be shielded, or removed, as needed.

- Flushing of Elevator Pit

  It is planned to flush the elevator shaft walls from the top to a few feet below the 305' level. It is planned to install a nozzle at the top of the shaft that will spray water (initially borated, and then deborated when available) at a rate of 25-gpm at 4000-6000 psi. A pump will be lowered into the shaft from the 305' level, to remove accumulated water from the bottom of the elevator pit.

- Cleaning Floor Drains

  The floor drains on the 305' level will be cleaned using a rotary grinding pipe cleaner with a flexible shaft. The head will be placed in the floor drain opening and will follow the drain down to the sump. While in operation, the drain will be continously flushed, so that all surface grinding and contaminants will flow down into the sump.

- Shielding the Seismic Gap and Penetrations at 305' Level

  This operation entails dose rate surveys around the gap and the penetrations, and shielding with lead blankets over these openings as needed for dose rate reduction.

- Decontamination and Shielding of Hot Spots

  A concerted effort will be made to locate and shield hot spots. The location of the hot spots will be done by using instrumented surveys, with TLD's, gamma ray spectrometers, and other instruments. Hot spots will be decontaminated or shielded. If they are located in pipes, it may be appropriate to flush the pipes and associated systems.

- Decontamination of Missile Shields

  After the missile shields have been moved by the polar crane, they will be scabbled as needed which will remove between 1/16th and 1/8th of an inch from affected surfaces. These surfaces will then be recoated. The dose rate from the shields will then be measured, and shielding will be added if needed.

- Shielding Reactor Head Service Structure

  To further reduce doses at the 347' level, the reactor head service structure will be shielded.

- Removal of Concrete and Paint from the 347' and 305' Level

  The floors on the 305' and 347' levels will be scabbled to reduce the dose from deeply absorbed contaminants in the concrete. Based upon experience with scabbling in the corridors of the auxiliary building, signficant reductions in the dose rate are anticipated. The experienced dose rate reduction in the auxiliary building was about 80%, with no significant increase in airborne contamination. Removal of paint is presently in the test and evaluation stage, with effect on dose reduction being investigated. It is anticipated that paint removal from the floor may significantly reduce the dose rate.

- Decontaminating Cable Trays at 305'

  The first step in this decontamination is to flush the cable trays with water, but should this prove insufficient, other methods will be tried, including steam cleaning using a hot steam-vacuum cleaner, or freon. Removal and partial (or total) replacement of the cable trays is considered to be a possible last resort.

- Decontamination of Equipment at the 347' Level

  On the 347' level, there are several large pieces of equipment, such as the RCS pump motor stand and the RCS pump storage stand. There are two methods under consideration to reduce the dose rate from these pieces of equipment. The first is to decontaminate the equipment in place, using proven techniques. The second is to decontaminate the equipment, in place, sufficiently so that workers will be able to cut them and remove the pieces from the containment for further decontamination. The method will be chosen based on ALARA considerations.

- Remote Flushing of the 282' Level

  It is planned to extend a remote nozzle through the seismic gap and other penetrations in the 305' level to flush the top of the 282' level.

- Remote Decontamination Technology Testing

  Over the next year robotic devices will be operated on the 282' level remotely from the 305' level. Some of the demonstrations planned include flushing, dispensing foams and gels, scabbling, and other decontamination procedures. This testing will be contigent upon available funding.

2.3 Decon Effectiveness Measurements

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

Exposures for the major Ongoing Containment Decontamination and dose reduction 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 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 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 Absorber (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. This source was calculated from continous air samples taken in the containment at a rate of 50 lpm. The concentration of various isotopes are given in Table 3-1.

It is assumed the particulate airborne concentration as given in Table 3-1 remains constant throughout any containment entry. It is also 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, and additional experience in the reactor building shows a marked reduction in airborne particulate activity. Also, it is assumed the containment purge is operated continuously.

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

The tritium concentration in the containment atmosphere was determined by using grab bubbler air samples which were previously described. These measurements give an average activity of 4.0E-7 µCi/cc. By using the same methodology as in the previous section, this results in a release of 1.50 E+2 Ci of tritium into the enviroment.

Using the value of 1.50 E+2 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 calculated release for tritium from the described activities is 1.50 E+2 Ci. This results in a calculated offsite dose of 5.84E-3 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)
Cs-134	2.78E-11
Cs-137	4.42E-10
Sr-90	7.70E-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.

Typically, an air sample taken at 50 LPM for 24 hours will normally have associated the following LLD's for a 1000 second count:

Cr-51	1.3E-12 μCi/cc
Mn-54	2.4E-13 μCi/cc
Fe-59	5.8E-13 μCi/cc
Co-58	2.4E-13 μCi/cc
Co-60	4.0E-13 μCi/cc
Zn-65	6.5E-13 μCi/cc
Ag-110m	1.9E-13 μCi/cc
Zr-95	3.9E-13 μCi/cc
Nb-95	2.2E-13 μCi/cc
Mo-99	1.0E-13 μCi/cc
Tc-99m	1.7E-13 μCi/cc
Ru-103	1.5E-13 μCi/cc
Ru-106	1.8E-12 μCi/cc
Sn-113	2.2E-13 μCi/cc
Cs-136	2.4E-13 μCi/cc
Ce-141	2.4E-13 μCi/cc
Ba-140	4.4E-13 μCi/cc
La-140	5.3E-13 μCi/cc
Sb-125	4.7E-13 μCi/cc
Ce-144	1.1E-12 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.

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 4.4E-15 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 3-3

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	2.78E-11	1.035E-5	3.280E-7	3.924E-3
Cs-137	4.42E-10	1.645E-4	5.216E-6	5.534E-2
Sr-90	7.70E-11	2.865E-5	9.086E-7	2.150E-2
H-3	4.02E-7	1.496E-1	4.744E-0	1.743E-1

4.0 OCCUPATIONAL EXPOSURE

4.1 External Exposure

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:

The planned in-containment man-hours to support the ongoing containment decontamination and dose reduction are 3000.

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 535 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 0.15 R/hour for elevation 305' (1330 hours), 0.10 R/hour for elevation 347' (1130 hours), and 0.08 R/hour for the polar crane and dome (540 hours). The resultant man-rem estimates are 200 man-rem for elevation 305'; 113 man-rem for elevation 347'; and 43 man-rem for the polar crane and dome. This yields 356 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 535 man-rem has been selected to be used as the estimate for the next year of the Ongoing Containment Decontamination and dose reduction.

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. The average BZA concentrations of Cs-134, Cs-137 and Sr-90 are shown below.

Cs-134	Cs-137	Sr-90
2.0E-9 μCi/cc	2.9E-8 μCi/cc	2.5E-9 μCi/cc

Although airborne radioactivity will decrease as a result of the decontamination activities, the concentrations of these isotopes 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 have indicated uptakes which would result from exposures to a mean tritium airborne activity level of 4.02E-07 μCi/cc or 8.0 E-2 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 than 5 X 10E3 dpm/100cm^2.

2. Reduce loose surface beta-gamma contamination levels on overheads to less than 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 will be made 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 effective technique on the basis of decontamination effectiveness per unit effort or expenditure.

5.0 RADIOACTIVE WASTE MANAGEMENT

5.1 Solid Waste

Solid waste categories have been 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.

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

5.2 Liquid Waste

5.2.1 Waste Water

A maximum of 500,000 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 floor 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.

5.2.2 Aqueous Chemical Waste

Chemicals will be considered for large scale decontamination uses only as approved by the Site Engineering department. Chemical waste will be appropriately treated and processed for disposal in a manner that is in compliance with federal regulations. For example, acid wastes will be neutralized prior to solidification and disposal.

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.

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 Body Cooling Devices In Containment

In order to reduce the problems with heat stress on containment workers, body cooling devices will be used when required. Two types of body cooling devices are available: Vortex Cooling Suits and Ice Vests. The vortex cooling suits circulates and cools air with a vortex fan. When mobility is required, ice vests will be used.

The safety and health staff will determine when heat protection gear will be required. This decision will be based on such factors as ambient temperature, the type of work to be done, work duration, and the type of radiation protective equipment used.

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 AND DOSE REDUCTION 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 and dose reduction 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. While flushing with deborated water, the reactor vessel will be covered to prevent leakage and dilution of the RCS boron. Until approved by the NRC 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 and dose reduction 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 and dose reduction 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 activities are consistent with ALARA considerations, and

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

9.0 REFERENCES

1) Radiological and Safety Evaluation of ongoing Containment Building Decontamination Activities for TMI-2 Recovery, Rev. 0, transmitted by GPU Letter 4410-82-L-0007 of September 23, 1982.