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

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U.S. NUCLEAR REGULATORY COMMISSION

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

Enclosed for your approval is the Safety and Radiological Evaluation of the Containment Building Decontamination Experiment. This document represents the radiological and safety evaluation of activities to be performed in the containment as given in the TMI-2 "Containment Building Decontamination Experiment Purpose and Scope of Work" transmitted to you on October 30, 1981. Included are evaluations of the following:

a) Effluents to the environment.

b) Occupational exposures.

c) Radioactive waste management.

d) Safety evaluation (10CFR50.59).

Based upon these evaluations GPU concludes 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 perform the decontamination experiment are consistent with ALARA considerations.

3. The decontamination experiment does not constitute an unreviewed safety question as defined by 10CFR50.59.

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Accordingly, we request that you approve the Radiological and Safety Evaluation, as well as the Scope, and authorize performance of the

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Decontamination Experiment upon your verification that requisite administrative controls are in place.

Sincerely,

J. J. Barton

J. J. Barton
Acting Director, TMI-2

JJB:JEL:djb

Enclosure

cc: L. H. Barrett, Deputy Program Director

RADIOLOGICAL AND SAFETY

EVALUATION OF THE CONTAINMENT BUILDING

DECONTAMINATION

EXPERIMENT

FOR

TMI-2 RECOVERY

Revision 0
November, 1981

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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) 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. That portion of tritium which is assumed to evaporate and become airborne from both the water in the sump and that used for the decontamination experiment 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, 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} $\mu Ci/cc$ 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 then 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.

2.1.2 Tritium Release

Measurements of the tritium concentration in the containment atmosphere, as measured by HPR-227, 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, one can draw the conclusion 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

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

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	Dose (mrem) 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.

3.0 OCCUPATIONAL EXPOSURE

3.1 External Exposures

Individuals entering the reactor building will be monitored for external exposure in accordance with GPU Radiological Control Procedures (RCP) to ensure personnel exposures are maintained within 10 CFR 20 dose equivalent guidelines. Administrative control points in accordance with GPU RCP 4015 will be used. Skin doses due to beta radiation will not be limiting based upon data from reactor building entries to date. Extremity monitoring will be performed in accordance with existing procedures. In addition, the planning and implementation of the decontamination experiment will be conducted to maintain exposures "as low as is reasonably achievable" (ALARA).

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, and the installation of a man-lift to provide access to the polar crane and elevation 347' from elevation 305'.

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 may be taken unless deemed unnecessary by bioassay, engineering judgement, or other substantive basis.

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. The section is divided into activities that reduce the radiation field that personnel are exposed to, and that reduce the time that must be spent in the radiation field to accomplish a task.

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

3.3.1 Reduction of the Radiation Field

Under some conditions, the installation of temporary shielding may be used to reduce the occupational exposure.

3.3.2 Reduction of Time Exposure

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 installating a new system in the containment to accomplish the same function.

4.0 RADIOACTIVE WASTE MANAGEMENT

4.1 Solid Waste

The decontamination experiment is expected to generate compactible trash in the form of plastic bags, booties, etc. Also, as a result of general area clean-up, non-compactible waste will be collected. Both the compactible and noncompactible waste will be removed from the containment and 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.

5.0 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.

6.0 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.