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January 18, 1985

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Equipment Hatch Removal Safety Evaluation Report

Attached for your review and approval is the Safety Evaluation Report (SER) for temporary removal and subsequent replacement of the Reactor Building equipment hatch, under specified conditions, to facilitate movement of heavy equipment into and/or out of the Reactor Building.

This SER concludes that the removal and reinstallation of the original equipment hatch can be accomplished without undue risk to the health and safety of the public.

As discussed in Section 4.0 of the SER, GPU Nuclear will submit a Technical Specification Change Request (TSCR) to incorporate limiting conditions for operation of the equipment hatch and modify the criteria relevant to reinstallation of the original equipment hatch, if replacement to less than the original configuration is desired.

The ability to utilize the equipment hatch will significantly enhance the preparation for starting defueling. Without this provision, several pieces of large defueling equipment will need to be disassembled and reassembled inside the Reactor Building. Such a reassembly requirement will result in increased radiation exposure to workers, both due to the reassembly and to additional operational checkouts. It will also result in incremental delays in the schedule.

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w/check $150.00
00014141

Pursuant to the requirements of 10 CFR 170, enclosed is a check for $150.00 for the application fee required for review of this submittal.

Sincerely,

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

FRS/RBS/jep

Attachment: (GPU Nuclear Check No. 00014141)

cc: Deputy Program Director - TMI Program Office, Dr. W. D. Travers

GPU Nuclear

SER 15737-2-G07-107 REV. 3

ISSUE DATE January 15, 1985

☒ ITS

☐ NSR

☐ NITS

TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

Removal of the

Equipment Hatch

COG ENG C. L. R.d DATE 12/7/84

RTR Edward T. Smith DATE 12/7/84

COG ENG MGR. C. L. R.d for RLR DATE 12/7/84

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
3	1/15/85	Revised and Reissued For Use	BJL	RJR	ETS	aR
2	1/8/85	Revised and Reissued For Use	RJR	D	ETS	WR
1	12/28/84	Revised and Reissued For Use	RJR	RLK	D	CLR
0	12/7/84	Issued for Use	RJR	D	ETS	CLR
				CHIEF ENGINEER N/A		

RB Date

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EPD-33611 10/84

Rev.	SUMMARY OF CHANGE
1	Deleted "for short periods of time" from Section 1.0; added requirement that both trains of purge system be operable when hatch is open; revised Section 4.0; added transmittal letter information to Section 5.0; and made minor word changes in Sections 2.4 and 3.2.
2	Revised descriptions of reactor building purge system operation.
3	Revised to delete references to replacement equipment hatch.

SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.
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BECHTEL

REVISION STATUS SHEET

DOCUMENT TITLE: Safety Evaluation Report for Removal of the Equipment Hatch

JOB 15737		REV.
SPEC NO. 2-G07-107		3
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GPD-33000-T 10/83

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1.0 PURPOSE AND SCOPE

The purpose of this Safety Evaluation Report (SER) is to demonstrate that the reactor building equipment hatch may be removed, under certain conditions, without undue risk to the health and safety of the public. Removal of the hatch will permit transport of large components needed to support defueling. Using this transport path will reduce the need to assemble equipment within the reactor building and, hence, reduce the exposure of workers to radiation. Further, this Safety Evaluation Report shows that the equipment hatch leak tightness requirements for the existing and future plant conditions, i.e., vented primary system, low primary coolant temperature, low decay heat, etc., are significantly less than those required to support reactor operation. Following hatch removal and material transport into containment, the equipment hatch opening will be closed by the existing hatch cover. 3

2.0 GENERAL DISCUSSION

2.1 Basis for Equipment Hatch Removal

The movement of equipment into and out of the reactor building is presently accomplished through the two personnel airlocks. Large equipment is disassembled outside of the reactor building into pieces small enough to be carried through the airlock, and reassembled once inside the reactor building. Therefore, large equipment designed for use inside the reactor building is modular in design with each modular piece compatible with being carried through the personnel airlock into the reactor building where all of the pieces will be assembled.

It will soon become necessary to move many large pieces of equipment essential for defueling into the reactor building. These items include:

1. Rotating Work Platform and Support Structure.
2. The Shielded Work Platform.
3. The Defueling Water Cleanup System components.
4. The dam for the deep end of the refueling canal.
5. The fuel storage racks (in canal).
6. The defueling canister/tool racks (in vessel).
7. The reactor building service crane.
8. Canister handling bridge trolley, including the canister transfer shield.
9. Various manual and automated defueling tools.

The movement of these items through the personnel airlock, and the subsequent assembly and testing of these items inside the reactor building, are major work evolutions with significant impact on radiation exposure, scheduling and resources allocation.

If the equipment hatch is removed to permit the transport of items, such as those listed above, into the reactor building, the items could be assembled and tested outside of the reactor building and brought in intact. This would eliminate much of the in containment assembly and testing required for the modular approach. The benefits of this are several fold:

a. Fewer Entries into the Reactor Building

The equipment hatch is large enough to permit assembled components to be brought into the reactor building. Therefore, one entry into the reactor building can replace numerous entries by workers carrying the pieces. Hence, radiation exposure associated with the installation of the equipment can be reduced.

b. Reduction in the Number of In-Containment Activities

The equipment brought in through the equipment hatch will be assembled so that minimum or no time must be spent in the building for assembly of the final pieces.

Startup testing and other pre-operational tests can be performed on equipment outside of the reactor building. Problems can be found and remedied before the equipment is brought into the building.

Also, this reduction in the number of activities will reduce delays resulting from space restrictions inside the reactor building. These activities could be performed in non-radioactive areas which would lower the radiation exposures for each activity.

c. Improved Worker Efficiency

Assembling major equipment outside of the reactor building will permit conventional construction practices in a non-contaminated environment. Thus, many evolutions such as welding, power tool operation, and rigging heavy lifts can be performed unencumbered by personnel protective clothing, respiratory protection, or the heat stress associated with working in the reactor building.

d. Simplified Equipment Design

The equipment design will be less complex because it will not have to be designed in modular form to permit transfer into the reactor building through a personnel air lock.

All of the above result in lower personnel radiation exposures, increased productivity, improved scheduling and optimized resource utilization.

2.2 Removal/Reinstallation of the Equipment Hatch

The equipment hatch is located in the southwest quadrant of the reactor building. It is a 23 ft. diameter penetration in the reactor building wall and is provided to permit the movement of large objects into and out of the reactor building during an outage. A removable personnel air lock (air lock #1) is incorporated into the equipment hatch. Both the hatch and the air lock are double gasketted and with the equipment hatch bolted to steel flanges in the building. The seal is designed to withstand the effects of the original design basis accidents for the plant. Figure 1 shows a sectional view of the equipment hatch, the personnel air lock, and the missile shield enclosure.

Prior to removal of the equipment hatch, the Containment Air Control Envelope (CACE) will be completed. The CACE provides an environmental barrier around the equipment hatch. The CACE serves as a collection point for monitoring any airborne release from the equipment hatch when the hatch is closed and assists in maintaining temperature limits (50°F minimum) inside containment when the personnel airlock or equipment hatch are open.

Prior to removing the equipment hatch, both trains of the Reactor Building purge system will be verified operable.

Prior to removing the equipment hatch, the inside surface of the equipment hatch and the personnel air lock will be decontaminated. This will minimize the spread of contamination when the personnel air lock and the equipment hatch are removed. The procedure for removing the equipment hatch requires that the personnel airlock be removed first. The airlock assembly will be withdrawn intact utilizing the monorail installed in the missile shield enclosure. The 9 ft. outside diameter, 12 ft. - 6 inches long airlock weighs 15 tons and is provided with lifting lugs to facilitate its removal. Once the personnel air lock is removed, the equipment hatch can be opened. The equipment hatch is 24 ft. - 8 inches outside diameter and weighs 20.5 tons. It can be removed using the installed monorail or other lifting equipment. Once removed the equipment hatch and the personnel airlock will be stored on site in an area suitable for staging contaminated material and protected against deterioration.

While the hatch is removed, a fire watch and security guard will be stationed at the hatch opening to ensure the integrity of the containment fire barrier and control personnel access, respectively.

Upon completion of the tasks which required the equipment hatch to be removed, the existing equipment hatch and personnel airlock will be replaced. This will be done to restore the reactor building integrity.

It will be replaced using the existing bolts and existing or replacement gaskets, but it will only be resealed to the requirements consistent with the current conditions in the building. Reactor building integrity will be assured by:

- Inspecting the seating surfaces before reinstalling the hatch.
- Inspecting the gaskets before reinstalling the hatch and replacing them as necessary.
- Installing the hatch and airlock in accordance with existing or new procedures.

2.3 Operational Restraints 3

The following restraints will be imposed during all times that the equipment hatch is removed for the movement of equipment into or out of the reactor 3
building.

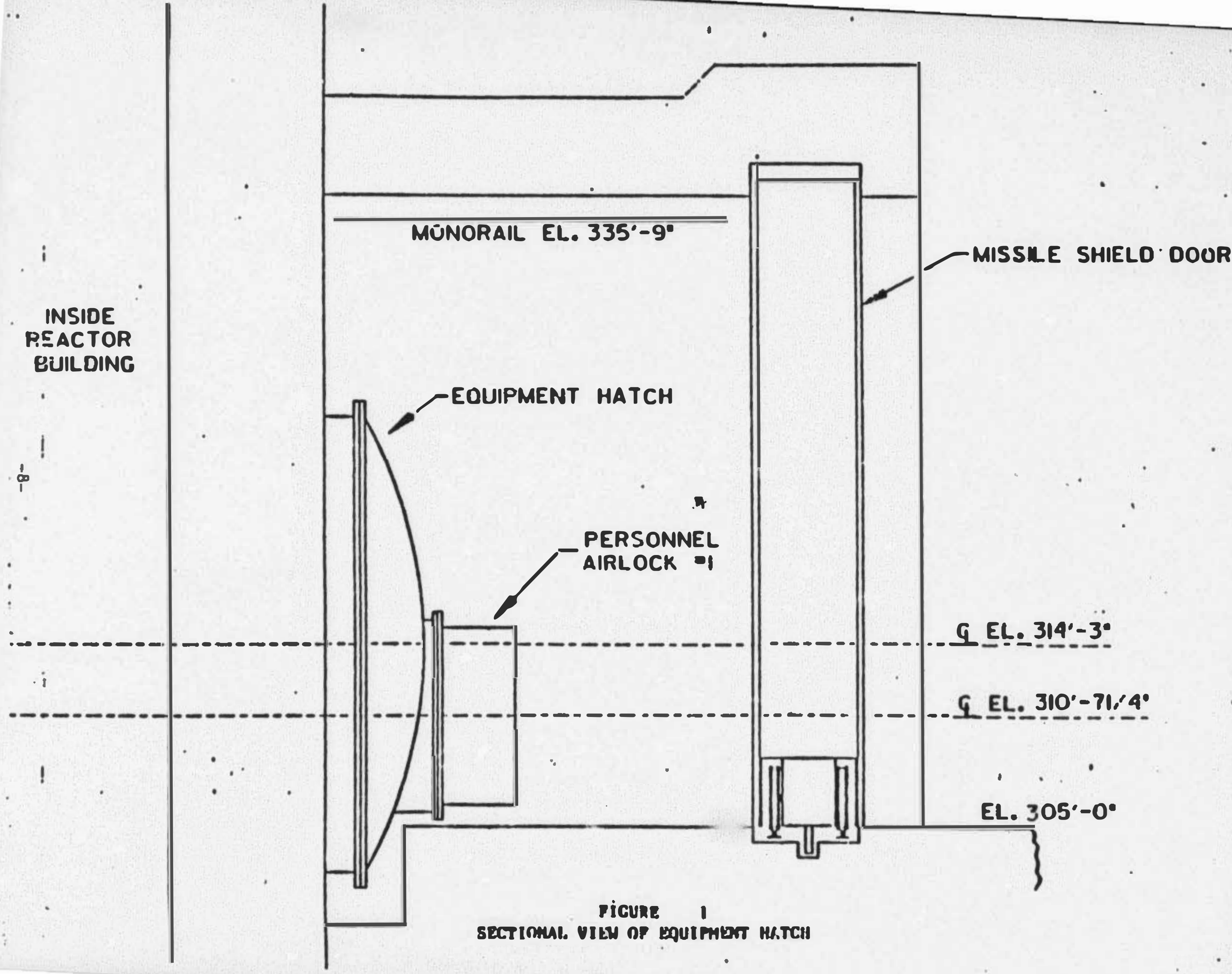

FIGURE 1
SECTIONAL VIEW OF EQUIPMENT HATCH

- No core alternations will be performed. Core alterations are defined as the movement or manipulations of any reactor component (including fuel) within the reactor pressure vessel with the vessel head removed and fuel in the vessel.
- Movement of defueling canisters containing fuel will be prohibited in the reactor building.
- Movement of loads over the reactor vessel, the incore seal table, the deep end of the refueling canal when canisters with fuel are located there, and the northwest quadrant of the "A" D-ring will be prohibited.
- At least one train of the reactor building purge system will be operated.
- The equipment access door of the CACE will be maintained closed whenever practicable.
- No decontamination operation will be performed.

3.0 ASSESSMENT OF SAFETY ISSUES

3.1 Identification of Safety Issues

The safety implications of removing the equipment hatch are related to the loss of the reactor building integrity. Due to the size of the equipment hatch, removing and replacing it takes longer (hours not minutes) than opening and closing both doors of the personnel airlock. Hence, upset conditions occurring when the equipment hatch is off cannot rely on rapidly reclosing the hatch in order to provide to the public the benefit of a fully closed containment structure.

The safety issues associated with the hatch being open are: 3

- Releases of radioactivity during normal activities.
- Releases of radioactivity resulting from off-normal events.
- Consequences of natural phenomena.
- Consequences of man made events.
- Assurance of reactor building integrity on re-installation of the hatch. 3
- Definition of operational restraints.

In addition to the safety issues identified above, a significant item associated with opening of the equipment hatch is the impact on the radiation exposure to workers. 3

All of the safety issues and the radiological impact on the workers are discussed in the subsequent sections.

3.2 Release of Radioactivity

To assess the acceptability of leaving the equipment hatch open, three release scenarios have been evaluated. The first scenario is the normal case when radioactivity release will be through the containment purge system filters and is identified as the normal release case. The second scenario postulates a short-term complete release of all airborne activity from the containment based on the activity levels associated with normal recovery mode operations and is identified as the puff release case. The third release scenario is the release from an accident inside containment which would cause a rapid increase of containment airborne radioactivity and an associated increase in the radioactive gaseous effluent from the containment building. This scenario is identified as the accident release case.

3.2.1 Normal Release Case

This case addresses the releases for the normal equipment batch removal
mode. Activities inside containment are restricted in accordance with
section 2.3 and the containment purge exhaust continues to operate (the |3
purge system includes two 100% capacity exhaust fan/filter trains).
The CACE equipment access door will normally be closed but may be
opened for short periods of time in this mode.

The purge system will maintain the containment at a negative pressure
(except when containment pressure is equalized with atmospheric
pressure to permit opening of the hatch) and thus prevent exfiltration |3
of containment air though the open hatch. Local air currents around
the hatch may cause direct release of very small amounts of containment
air particularly when the CACE equipment access door is open. The
amount of release through the open hatch will be further limited by
minimizing the length of time that the CACE door will be open.

The level of total release in this mode is expected to be less than the
normal releases from containment when the hatch is closed due to the
restrictions imposed in Section 2.3 and the continued operation of the |3
containment purge system.

3.2.2 Puff Release Case

This scenario postulates the release of all containment airborne activity within a short period of time, taking no credit for the CACE or for containment purge system operation. The activity levels used for this assessment are based on actual measurements of airborne activity inside containment during normal recovery operation. This type of assessment is presented in Recovery Operations Plan Change Request (ROPCR) No. 18, Reference 9, for opening of both personnel airlock doors. The release levels presented in ROPCR No. 18 are also applicable to equipment hatch removal and demonstrate that the releases associated with this scenario are acceptable for the following reasons:

1. Normal containment airborne activity levels at the time of hatch removal are expected to be lower than those used as a basis for the ROPCR No. 18 assessment due to ongoing decontamination work inside containment. The containment airborne activity levels used for the ROPCR No. 18 assessment are based on actual measurements taken in 1982. Actual activity levels measured in late summer 1984 indicated that levels are substantially lower today.

2. No further significant releases are expected to occur after the
complete 30-minute releases postulated in ROPCR No. 18 due to
the operational restrictions noted in Section 2.3 of this SER. |3

The actual release of activity when the hatch is open will be significantly reduced by the operation of the containment purge system in the exhaust mode which will minimize outward airflow at the hatch. In addition, the CACE equipment access door will be maintained closed whenever possible while the containment hatch is removed further minimizing the potential for significant direct release of airborne activity. 3

3.2.3 Accident Release Case

Of the possible in-containment accidents, only a large fire could cause a rapid increase in the containment airborne radioactivity and also be accompanied with an energy release which could develop a driving force to expel the airborne contaminants from the containment.

Inside containment there are two designated waste storage areas. One is located on the 305' elevation near the personnel air lock and the other is located on the 347' elevation adjacent to the enclosed stairwell. The bounding accident release case is the postulated fire of the storage area located on the 305' elevation. The dose assessment is based on the instantaneous release of airborne contaminants resulting from an all-consuming fire in this storage area. Such an all-consuming fire assumes no actions are taken to control and put out the fire.

An estimate has been made of the maximum waste that could be in the storage area at any one time and the radioactive content of that waste. Table 1 gives the maximum waste and its isotopic inventory in the storage area. The amount of respirable airborne release from this storage area fire is assumed to be 0.1% of the inventory in the storage area. Table 10 of Appendix B of Reference 4 and page 8-61 of Reference 5 are cited as the bases for this release fraction.

To determine the dose to the maximally exposed individual from the postulated fire inside containment the release is assumed instantaneous and the passing cloud of radioactivity travels to the nearest site boundary. The radiological consequence of this accidental release is dependent on the meteorological conditions present at the time of release. A conservative approach is taken which assumes that the 5 percentile 0-1 hour atmospheric dispersion factor at the nearest site boundary exists at the time of release. This dispersion factor is taken from Appendix 2D of the TMI- Unit 2 FSAR and has a value of 6.1E-4 sec/M^3.

The dose assessment is based on the inhalation pathway using the methodology and dose factors found in Reference 2. The breathing rates presented in Table E-5 of Reference 2 are increased by a factor equal to the ratio of the breathing rate given in Reference 6 to the adult breathing rate given in Table E-5 of Reference 2. The doses to the various organs for the different age groups are calculated to determine the maximum dose. The most restrictive organ is the teenager's bone with a dose of 0.027 mrem.

These results are a small fraction of the limits specified in 10CFR100 for releases resulting from accidents.

Table 1

Inventory of Storage Areas Inside Containment

Quantity of Waste	6,000 pounds
Inventory of Waste	
Cs-137	0.201 curies
Cs-134	0.008 curies
Sr-90	0.008 curies

3.3 Radiological Assessment

Dose rates outside the equipment hatch are very low. Measurements taken approximately 10 ft. from the hatch with both airlock doors open have shown the dose rate to be < 1 mrem/hr. This will provide an area of low radiation levels for staging material and pre-assembling components. With the equipment hatch removed, it has been estimated by radiological controls personnel that the dose rate at 10 ft. from the hatch will be approximately 10 mrem/hr.

As discussed in Section 2, removing the equipment hatch will result in reduced occupational radiation exposure associated with defueling. An estimate of the reduction in occupational exposure has been made by the Technical Assistance and Advisory Group (TA&G), Reference 7. The results of this estimate are presented in Table 2. As noted in Reference 5,

> All ALARA estimates are based on assumptions. This particular estimate is further hampered by the imprecise knowledge of the real scope and extent of work to be performed. For this reason, conservative assumptions were made to reduce the advantage of opening the equipment hatch. The amount of effort required to bring in pieces, to assemble components, and to startup and test systems in the reactor building have been minimized. Even so, the results indicate that the radiation exposure drops by a factor of six if the equipment hatch is used. While by no means definitive, the results of this scoping study ought to represent the minimum expected ratio between worker exposures with the equipment hatch open and with the equipment hatch closed. Actual savings are expected to be significantly larger.

Based on this, it is concluded that removal of the hatch will result in a significant reduction in occupational radiation exposure associated with defueling activites.

3.4 Natural Phenomena

3.4.1 Floods

The equipment hatch is 23' in diameter centered at elevation 314'-3". The elevation of the probable maximum flood (PMF) for TMI-2 is 308.5' with a predicted wave action of an additional 4' for a total flood level of 312.5'. If a flood of this severity occurs while the equipment hatch is open flood water could conceivably fill the containment basement.

The PMF is a low probability event that cannot occur without ample warning. No dams or large reservoirs exist immediately upstream of the site which could, of and by themselves, generate that PMF even if all failed at once. The PMF can only occur as the result of a heavy, prolonged rain storm. In 1972, hurricane Agnes, which caused the highest flood water elevations ever recorded for the Susquehanna River, resulted in a flood elevation of approximately 301' which is four feet

TABLE 2

ALARA ESTIMATE FOR EQUIPMENT HANDLING IN THE REACTOR BUILDING

	No. of Entries (1)	(2)	Average Dose Rate (mr/hr) in R.B. El. 305'-0" (3)	(4)	Average Dose Rate (mr/hr) in R.B. for Construction and/or Testing		Man Hours per Entry (5)	(6)	Man Hours for Construction		Man Hours for Testing		Average Entry Radiation Man-Rem Note 1	Note 2	Average Area Radiation Man-Rem due to Construction and Testing		Total Man-Rem	
	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2
To Support Plenum Removal																		
• Work Platform	50	1	175	220	50*	N.A.	1	2	50	N.A.	N.A.	N.A.	8.75	0.44	2.75	N.A.	11.50	0.44
To Support DHCS																		
• Piping Manifolds	12	1	175	220	50*	N.A.	2	2	12	N.A.	24	8	4.20	0.44	1.97	0.46	6.17	0.90
To Support Dry Defueling																		
• Dam for Deep End of Canal	1	1	175	220	50*	N.A.	4	2	1	N.A.	N.A.	N.A.	0.70	0.44	0.08	N.A.	0.78	0.44
• Support Structure	12	1	175	220	50*	N.A.	1	2	12	N.A.	N.A.	N.A.	2.10	0.44	0.65	N.A.	2.75	0.44
• Fuel Racks in Canal	10	1	175	220	50*	N.A.	4	4	32	N.A.	N.A.	N.A.	7.00	0.88	1.77	N.A.	8.77	0.88
• Canister/Tool Racks Vessel	6	1	175	220	50*	N.A.	1	2	6	N.A.	N.A.	N.A.	1.05	0.44	0.33	N.A.	1.38	0.44
• Mods for Main Fuel Handling Bridge	6	1	175	220	50*	N.A.	1	2	6	N.A.	4	4	1.05	0.44	0.56	0.23	1.61	0.67
• Rotating Work Platform	80	1	175	220	50*	N.A.	1	2	80	N.A.	40	N.A.	8.99	0.44	6.56	N.A.	15.55	0.44
To Support Complete Program																		
• Gantry Crane	5	1	175	220	50*	N.A.	2	2	8	N.A.	24	N.A.	1.75	0.44	1.59	N.A.	3.34	0.44
• Transfer Bell	5	1	175	220	50*	N.A.	2	2	8	N.A.	6	N.A.	1.75	0.44	0.61	N.A.	2.36	0.44
• RPV Gasket	1	1	175	220	50*	N.A.	1	1	N.A.	N.A.	N.A.	N.A.	0.18	0.22	N.A.	N.A.	0.18	0.22
• 50 Long Handled Tools	50	1	175	220	50*	N.A.	0.5	2	N.A.	N.A.	N.A.	N.A.	4.38	0.44	N.A.	N.A.	4.38	0.44
Totals							21.5	22	209	N.A.	98	12	41.99	5.94	16.87	0.69	58.22	6.12

Case 1 - ALARA for equipment handling in the reactor building using personnel air lock entry.
Case 2 - ALARA for equipment handling in the reactor building using equipment hatch entry.

* Exposure to workers during construction and testing is 50 mR/hr.
The dose for each entry and exit per worker is 14 mR in addition to stay time.

Note 1 - Column 1 x Column 3 x Column 5
Note 2 - Column 2 x Column 4 x Column 6

below that level required to flood the hatch opening (305'). Due to the nature of rain-induced flooding, ample warning will exist to assure that the equipment hatch can be replaced prior to the onset of a flood that can affect the reactor building. The existing procedures already identify three stages of flood alert: ALERT (36-hour forecast of 640,000 cfs or greater), EMERGENCY CLOSURE (36-hour forecast of 940,000 cfs or greater) and SHUTDOWN (301 foot water level at River Water intake structure corresponding to a river water flow of 950,000 cfs).

Prior to reaching those levels, an initial warning will be received from the Federal-State River Forecast Center when the 36 hour river flow forecast for Harrisburg indicates 350,000 cfs. Procedures will require that the equipment hatch will be replaced when the 350,000 cfs warning is received. This will provide assurance that adequate time will be available to close the hatch before the river level reaches the bottom of the hatch, thereby preventing flood water from entering the reactor building. |3

3.4.2 Missile

Currently, the moveable missile shield is retracted. The equipment hatch provides limited protection against missiles. However, the secondary shield wall is inside of the equipment hatch opening and will prevent any conceivable missile from penetrating into the area where it could affect the reactor coolant system. The secondary shield wall is a 5000 psi concrete seismic category I structure 4'-6" thick extending from elevation 282'-6" to 374'-4" and is as strong as the missile shield. Hence, the missile protection is not dependent on the presence of the equipment hatch and will not be degraded by its removal.

3.4.3 High Winds

The equipment hatch will be enclosed by the CACE. From the TMI-2 FSAR, the design wind velocity, based on the 100-year recurrence interval, is 80 miles per hour at 30 feet above grade. The CACE is designed to withstand this condition. Thus high winds will not result in an unacceptable increase in releases of radioactivity to the environment.

3.4.4 Seismic Event

It is noted that the equipment hatch serves no structural function so that the seismic integrity of the reactor building is not affected by its removal. In addition, the seismic capabilities of the engineering safeguards are not dependent on the presence of the equipment hatch, although reactor building integrity would not be maintained.

Based on operating within the limitations discussed in Section 2.3, the |3
amount of radioactivity that would be released to the environment and
the resulting doses are bounded by the analyses discussed in Section
3.2. This conclusion is reached based on no activities being performed
that will significantly increase the airborne radiation levels in the
reactor building and the assumption of no causal effect from a seismic
event on the amount of airborne activity.

3.5 Man Made Events

The only man made event that removal of the equipment hatch will be impacted
by is an airplane crash. Section 3.5.3.3 of the TMI 2 FSAR shows that the
probabilty of an airplane crash which damages safety related buildings, is a
very low probability event. An airplane crash which impacts on the equipment |3
hatch is judged to be so unlikely as to be deemed incredible.

3.6 Operational Considerations

Operational considerations associated with the removal of the equipment hatch
include the operational restraints listed in Section 2.3 and only requiring
that the equipment hatch be installed to withstand only a 2 psi containment |3
pressure. The bases for these are discussed below.

When the equipment hatch is removed it cannot be reinstalled quickly in the
event that containment integrity must be reestablished. Prohibiting
activities that involve the handling of fuel will prevent the possible spread |3
of fuel to environment in the event of an accident requiring the establishment
of containment integrity. This is also true for handling of canisters filled
with fuel and the handling of heavy loads over areas which could result in
disturbance of fuel or result in draining the RCS below the RV nozzles.
Draining the RCS to RV nozzles has been judged acceptable based on evaluations
presented in Reference 3.

Having the reactor building air handling system operating as described in
Section 2.3 will ensure that direct release of airborne activity through the |3
equipment hatch will be minimized.

The prohibition on decontamination activities in containment will ensure that
airborne activity levels will not be increased when the hatch is open beyond
the levels used in the referenced analysis.

When reinstalled after removal the hatch will only be capable of withstanding
a 2 psi reactor building pressure. This pressure exceeds the design pressure |3
of the most limiting containment penetration. The justification for
containment penetration design pressure is provided in Reference 8.

4.0 10CFR50.59 EVALUATION

10CFR50, 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 possibility 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.

It has been demonstrated in Section 3 of this report that removal of the equipment hatch does not result in increasing the possibility or consequences of an accident, or create a different type of accident than that which was evaluated for opening the two doors of the personnel airlock. Removal of the equipment hatch will require a modification to the technical specifications if replacement to less than the original configuration is desired (including re-installation of the existing hatch using fewer closure bolts), however, the margin of safety as defined in the current bases will not be reduced. | 3

Therefore, removal of the equipment hatch as described in this SER is not an unreviewed safety question and can be accomplished without undue risk to the health and safety of the public.

5.0 References

1. TMI-Unit 1 Offsite Dose Calculation Manual, Rev. 4, 8/31/84.

2. NRC Regulatory Guide 1.109, Rev. 1, October, 1977.

3. GPUN letter 4410-84-L-0014, dated March 9, 1984, from B. K. Kanga to B. J. Snyder, transmitting "Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head," Revision 5, February 1984.

4. "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," USAEC, December, 1972.

5. "Final Programmatic Environmental Impact Statement for TMI-Unit 2," NUREG-0683, March, 1981.

6. NRC Regulatory Guide 1.4, Rev. 2, June, 1974.

7. Ninth Report of the Technical Assistance and Advisory Group (TAAG) for the Period April 1, 1984 to August 1, 1984.

8. GPUN letter LL281-0191, December 4, 1981, from G. Hovey to Snyder, "Design Pressure for Containment and Future Mechanical and Electrical Penetration Modifications," December 4, 1981.

9. GPUN letter 4410-82-L-0023 of October 6, 1982 to USNRC, from B. K. Kanga to B. J. Snyder, "Recovery Operations Change Request No. 18."