AN ANALYSIS
OF A
FUEL HANDLING ACCIDENT
IN THE
REACTOR BUILDING

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

FOR

METROPOLITAN EDISON COMPANY

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1.0 INTRODUCTION

This report presents the results of an analysis of a postulated fuel handling accident in the Reactor Building of Three Mile Island Nuclear Station, Unit 1. The following sections describe the relevant ventilation and monitoring equipment and the analysis of the radiological consequences of the postulated accident.

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2.0 REACTOR BUILDING VENTILATION DURING REFUELING OPERATIONS

The following HVAC equipment is used during refueling:

1. Two of the thre. Reactor Bu [illegible] oling units (AH-E-1A, 1B, 1C).

2. Both o[illegible] the two oper [illegible]ng floor supply fans (AH-E-3A, 3B).

In addition, the purge supply and the purge exhaust system (AH-E-6[illegible], 6B, 7A, 7B) can, under certain circumstances, also be used during refueling operations.

Operation of two Reactor Building cooling units results in the cooling and circulation of 216,000 cfm. The cooled air is discharged along the north wall at elevation 281'. After circulating through the Reactor Building, this air is gathered along the north, south, and west portions of the Reactor Building at elevation 416'-6" and is directed back to the Reactor Building cooling units.

Operation of the two operating floor supply fans results in the transfer of 90,000 cfm of cooled air from the 281' elevation to distribution ductwork at 375'-0". The air is transferred from the southwest and northeast quadrants of the lower elevation and is distributed along the southeast and northeast elevation at 375'-0".

In order to operate the purge supply and exhaust system, Environmental Technical Specification 2.3.2 requires that:

1. The Reactor Building purge exhaust monitor RM-A9 shall be operable.

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2. The purge exhaust valves AH-VlA and AH-VlB shall be operable.

3. The valves AH-VlA and AH-VlB shall be interlocked to close on receipt of a high radiation signal from the Reactor Building exhaust monitor RM-A9.

Operation of the purge supply and exhaust system results in the supply of 50,000 cfm of outside air to the northwest quadrant of the Reactor Building at elevation 281' and exhaust of the same quantity from the southwest quadrant at elevation 317'. This exhaust is directed to a filter plenum containing roughing, HEPA, and charcoal filters as depicted in Figure 9-19 of the FSAR. From the filters, the exhaust enters the main unit vent.

The purge exhaust filter plenum is nominally sized to filter 50,000 cfm. The plenum houses 56 roughing filters, 56 HEPA filters, and 168 charcoal filter trays. The HEPA filters are in accordance with MIL-F-51079 and are nominally sized at 24x24x12 inches deep. The charcoal filter trays are nominally sized at 24½x26½x6½ inches. The trays are 304 stainless steel construction and the charcoal media is activated coconut shell, impregnated with potassium iodide, type MSA 85851.

The installed filters were initially field tested with DOP smoke to determine HEPA filter leakage and with Freon 112 to determine bypass leakage of the charcoal filters. In addition, they are periodically tested as required by Technical Specification 4.14.1:

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At intervals not exceeding refueling interval, leakage tests using DOP on the HEPA filter and Freon-112 (or equivalent) on the charcoal unit shall be performed. Removal of 99.5 percent DOP by the HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by the charcoal adsorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of the filtration units or of the housing.

It is anticipated that contaminants released during refueling at elevation 346' would be gathered by the ductwork at 416'-6" and returned to the 281' elevation. At this lower elevation, the 216,000 cfm return air flow would be mixed and diluted with the purge supply air at the same elevation. A portion of this mixture would be exhausted by the purge exhaust, and at elevation 308' the remaining air would travel back to the 416'-6" elevation for recirculation.

The 216,000 cfm return air flow to the ductwork at elevation 416'-6" represents approximately 8.7 air changes per hour for the volume above elevation 346'-0".

The 216,000 fan return and the 50,000 cfm purge supply to the 281' elevation represents approximately 46 air changes per hour for the volume above elevation 281'. Thus, contaminants returned to this

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level are rapidly transferred to upper elevations and are diluted by the purge supply in a ratio of approximately 20%.

The Reactor Building cooling units and the purge supply and exhaust valves are safety related equipment and can withstand LOCA conditions and the seismic event. All other HVAC equipment discussed above is designed to operate in the normal expected plant environment.

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3.0 REACTOR BUILDING RADIATION MONITORING

The radiation monitoring equipment available to detect and/or monitor the radioactivity release associated with a postulated fuel handling accident in the Reactor Building consists of area gamma detectors located inside the Reactor Building and fixed atmospheric monitors located outside the Reactor Building as described in Chapter 11 of the FSAR. Both sets of monitors are discussed below.

Area Gamma Monitors

Area gamma monitors located in the Reactor Building are as follows:

1. Reactor Building personnel access hatch (RM-G5)
2. Reactor Building fuel handling bridge no. 1 (RM-G6)
3. Reactor Building fuel handling bridge no. 2 (RM-G7)
4. Reactor Building high range (RM-G8)

Each of these monitors is equipped with an ionization chamber detector housed in a weather proof container and equipped with a control room controlled check source. The energy response of the detector is $\pm$10 percent for gamma radiation in the 80 Kev to 3 Mev range. An overall channel accuracy of $\pm$20 percent of actual radiation intensity is achieved.

Indication and alarm are provided both in the control room and in the area monitored. Each channel is capable of measuring radiation over a range of 8 decades. RM-G5, RM-G6 and RM-G7 have a range from 0.1 to 1×10^7 mR/hr. RM-G8 is desensitized by a lead shield to measure up to 1×10^6 R/hr. The response time of each of the above

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detectors varies with the intensity of the radiation to be measured from approximately 3 to 15 seconds. While bought as commercial grade equipment, this type of equipment has been found capable of withstanding seismic loads of 1g in the frequency range of 1 - 30 Hz.

In the unlikely event of a fuel handling accident inside the Reactor Building, radiation detectors RM-G6 and RM-G7 which are located on each of the refueling bridges and monitor gamma activity in the vicinity of the water surface are the most suitable to detect and alarm any excessive radiation level above their alert set points set at 1.5 R/hr and 0.3 R/hr, respectively, or their high alarm set points set at 2.5 R/hr and 0.75 R/hr, respectively. This would provide indication of this incident both locally and in the control room. If either monitor is inoperable, Technical Specification 3.8.1 requires that portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

Atmospheric Monitors

The Reactor Building atmosphere is monitored continuously for radioactivity by an atmospheric monitor (RM-A2) which is capable of detecting radioactive particulates, halogens, and noble gases with the following sensitivities:

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particulates	2.0×10^2 cpm/μCi/cc	(Sr-90)
halogens	1.2×10^9 cpm/min/μCi/cc	(I-131)
noble gases	4.0×10^7 cpm/μCi/cc	(Xe-133)

The primary function of RM-A2 is to detect and monitor primary coolant leakage. The monitor is designed to draw a sample of air from the Reactor Building and collect and monitor the build up of radioactive particulate using a moving filter, collect and monitor radioactive iodine using a charcoal cartridge, and instantaneously measure noble gases using a continuous gas monitor. RM-A2 is located outside the Reactor Building.

The Reactor Building purge exhaust is monitored for radioactive particulate, iodine and gas by a monitor (RM-A9) located downstream of the charcoal filter and prior to discharge to the environment. The location of the monitor has been selected to ensure the monitoring of a representative sample of the discharge. The operation of the monitor is similar to RM-A2 described above except that airborne particulate are collected on a fixed filter. The sensitivities of RM-A9 for detecting particulates, halogens, and noble gases are as follows:

particulates	1.8×10^{10} cpm/min/μCi/cc	(Sr-90)
halogens	1.4×10^9 cpm/min/μCi/cc	(I-131)
noble gases	3.9×10^7 cpm/μCi/cc	(Xe-133)

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In the unlikely event of a fuel handling accident inside the Reactor Building, excessive radiation above the following setpoints for RM-A2 and RM-A9

		Alert Setpoint (cpm)	High Alarm Setpoint (cpm)
RM-A2	particulates	1×10^5	2×10^5
	halogens	2×10^5	6×10^5
	noble gases	1×10^4	2×10^4
RM-A9	particulates*	1.4×10^4	9×10^5
	halogens	6.8×10^2	8.6×10^4
	noble gases	7.1×10^3	4.5×10^4

* Fixed filter - alarm will be dependent on rate of buildup.

will provide an alarm. In addition, an electrical interlock from the Reactor Building purge exhaust gas monitor (RM-A9) will automatically close the Reactor Building purge supply and exhaust isolation valves. These valves are designed to close in 5 seconds.

RM-A2 and RM-A9 are commercial grade and appropriately designed to operate in the normal expected plant environment. The equipment is rugged but not seismic qualified or designed to meet the single failure criteria. Failure of the radiation monitoring equipment, however, will not degrade the safety qualification of the Reactor Building isolation valves.

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4.0 RADIOLOGICAL CONSEQUENCES

The analysis of the postulated fuel handling accident in the Reactor Building is based on the following:

1. The accident is assumed to happen after the reactor has been shutdown for 72 hours. This is based on Technical Specification 3.8.10 which requires at least 72 hours between reactor shutdown and the removal of irradiated fuel. Radioactive decay of the core fission product inventory during this interval is taken into account.

2. All of the rods in one assembly are assumed to rupture as a result of the accident.

3. The assembly damaged is assumed to be the highest powered assembly in the core region to be discharged. Table 1 gives iodine and noble gas inventories for the core, an average assembly, and the maximum assembly based on the conservative assumption that the entire core is irradiated at full power for 930 days. The core inventories are based on Table 15A-2 of the Unit 2 FSAR. The inventories for an average assembly are determined by dividing the core inventories by the total number of assemblies in the core. The inventories in the maximum assembly are determined by applying a radial peaking factor of 1.7 to the inventories in an average assembly.

4. All of the activity in the clad gap in the damaged rods is released to the refueling water. This activity is based on Regulatory Guide 1.25 assumptions, i.e., 10 percent of the total

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noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident.

5. The iodine gap inventory composition is based on Regulatory Guide 1.25 assumptions, i.e., 99.75 percent inorganic species and 0.25 percent organic species.

6. The refueling water decontamination factors are based on Regulatory Guide 1.25 assumptions, i.e., 133 for inorganic iodine species and 1 for noble gases and organic iodine species.

7. The radioactive material that escapes from the refueling water is released from the building through the charcoal filters in the purge exhaust.

8. The iodine removal efficiencies for the purge exhaust filters are based on Regulatory Guide 1.25 assumptions, i.e., 90 percent for the inorganic iodine species and 70 percent for the organic iodine species.

9. No credit is taken for holdup in the Reactor Building. However, existing Technical Specifications 3.8.6, 3.8.7 and 3.8.9 which are as follows

   3.8.6 During the handling of irradiated fuel in the reactor building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall

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be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.9 The reactor building purge system, including the radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than one week prior to refueling operations.

provide assurance of automatic Reactor Building isolation in the event of a fuel handling accident in the Reactor Building. In addition, radiation monitors RM-G6 and RM-G7 which alarm any excessive radiation in the vicinity of the refueling water surface plus Technical Specification 3.8.5 which requires that direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place provide assurance that in the event of a fuel handling accident in the Reactor Building the control room operators would have sufficient information to initiate isolation of the Reactor Building.

10. Atmospheric diffusion is calculated using a 0 - 2 hour dispersion factor at the exclusion boundary of 6.1×10^{-4} sec/m^3. This value is based on Table 6.2-9c submitted in Amendment 48 to the FSAR for Unit 2.

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Isotopic releases to the atmosphere using these assumptions are summarized in Table 2. The resulting thyroid and whole body doses at the exclusion boundary are 25.4 and 0.65 Rem, respectively.

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5.0 CONCLUSIONS

Conservatively calculated exclusion boundary radiation exposures due to a postulated fuel handling accident in the Reactor Building are well within the guidelines of 10CFR100. Since the calculations were performed without taking credit for Reactor Building isolation, no changes to facility equipment or Technical Specifications have been considered. For the same reason, an evaluation of the consequences of the accident assuming a single failure was not performed.

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TABLE 1

FISSION PRODUCT INVENTORIES
FOR THE CORE, THE AVERAGE ASSEMBLY,
AND THE MAXIMUM ASSEMBLY

Activity (Curies)

Isotope	Core*	Average Assembly	Maximum Assembly**
Kr-85m	2.33 (+7)***	1.32 (+5)	2.24 (+5)
85	8.54 (+5)	4.82 (+3)	8.20 (+3)
87	4.27 (+7)	2.41 (+5)	4.10 (+5)
88	6.46 (+7)	3.65 (+5)	6.20 (+5)
Xe-131m	5.90 (+5)	3.33 (+3)	5.67 (+3)
133m	3.38 (+6)	1.91 (+4)	3.25 (+4)
133	1.40 (+8)	7.91 (+5)	1.34 (+6)
135m	3.69 (+7)	2.08 (+5)	3.54 (+5)
135	2.83 (+7)	1.60 (+5)	2.72 (+5)
I-131	6.96 (+7)	3.93 (+5)	6.68 (+5)
132	1.06 (+8)	5.99 (+5)	1.02 (+6)
133	1.56 (+8)	8.81 (+5)	1.50 (+6)
134	1.83 (+8)	1.03 (+6)	1.76 (+6)
135	1.42 (+8)	8.02 (+5)	1.36 (+6)

* Based on irradiation of the entire core at full power for 930 days.

** Based on a radial peaking factor of 1.7.

*** 2.33 (+7) = 2.33×10^{7}

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TABLE 2

RADIOACTIVE RELEASE FOR
THE POSTULATED FUEL HANDLING
ACCIDENT - REGULATORY GUIDE 1.25 ANALYSIS

Isotope	Activity Released (Curies)
Kr-83m	2.67 (-1)*
85	2.46 (+3)
Xe-131m	4.75 (+2)
133m	1.30 (+3)
133	9.03 (+4)
135	1.20 (+2)
I-131	7.74 (+1)
133	7.47
135	2.10 (+1)

* 2.67(-1) = 2.67 x 10^{-1}

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