U. S. NUCLEAR REGULATORY COMMISSION

Report No. 50-320/85-16

Docket No. 50-320

License No. DPR-73 Priority -- Category C

Licensee: GPU Nuclear Corporation

P.O. Box 480

Middletown, Pennsylvania 17057

Facility Name: Three Mile Island Nuclear Station, Unit 2

Inspection At: Middletown, Pennsylvania

Inspection Conducted: August 6, 1985 - September 6, 1985

Inspectors:	R. Cook, Senior Resident Inspector (TMI-2)	9/20/85 date signed
	T. Moslak, Resident Inspector (TMI-2)	9/20/85 date signed
	J. Bell, Senior Radiation Specialist	9/24/85 date signed
	D. Collins, Radiation Specialist	9/20/85 date signed
	L. Myers, Radiation Specialist	9/20/85 date signed
Approved By:	C. Cowgill, Chief, TMI-2 Project Section	9/24/85 date signed

Inspection Summary:
Areas Inspected: Routine safety inspection by site inspectors of plant operations (long term shutdown), including in-plant review of Licensee Event Reports, transfer of radioactive liquids, examination of welding on Canister Storage Modules, issuance of respiratory protection equipment, housekeeping of Reactor Building, and radioactive waste shipments. The inspection involved 317 inspector hours.
Results: A violation was identified in the issuance of respiratory protection equipment (paragraph 8.0). One worker was observed to be not fully alert at the entry point to the Reactor Building. This is the second time this has been observed in the past four months.

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DETAILS

1.0 Ongoing Recovery Operations

a. Routine Plant Operations

Inspections of the facility were conducted to assess compliance with the requirements of the Proposed Technical Specifications and Recovery Operations Plan in the following areas: licensee review of selected plant parameters for abnormal trends; plant status from a maintenance/modification viewpoint, including plant cleanliness, control of switching and tagging, and fire protection; licensee control of routine and special evolutions, including control room personnel awareness of these evolutions; control of documents, including log keeping practices; radiological controls, and security plan implementation.

Random inspections of the control room during regular and back shift hours were routinely conducted. The Shift Foreman's Log and selected portions of the Control Room Operator's Log were reviewed for the period August 6 through September 6, 1985. Other logs reviewed during the inspection period included the Submerged Demineralizer System (SDS) Operations Log, Radiological Controls Foreman's Log, and Auxiliary Operator's Daily Log Sheets.

Operability of components in systems required to be available for response to emergencies was reviewed to verify that they could perform their intended functions. The inspectors attended selected licensee planning meetings. Shift staffing for licensed operators, non-licensed personnel, and fire brigade members was observed.

No violations were identified.

2.0 Licensee Event Report (LER) Review

The inspector reviewed the LER's listed below which were submitted to the NRC TMI Program Office to verify that the details of the events were clearly reported, including the accuracy of the description of the causes and the adequacy of corrective actions. The inspector determined whether further information was required from the licensee, whether the event should be classified as an Abnormal Occurrence, whether generic implications were indicated, and whether the event warranted onsite followup.

-- LER 85-07, dated July 21, 1985, addressed a licensee identified condition in which a containment isolation valve, FS-V-639 had been routinely operated without using an NRC-approved procedure per the requirements of Technical Specifications 6.8.2. The period during which this condition occurred was April 13, 1984 through July 2, 1985. The cause is attributed to a clerical error during the conversion of procedure 4300-ADM-3240.1 to 4000-ADM-3240.01 (both NRC-approved) in which FS-V-639 was inadvertently deleted.

The inspector determined, by reviewing the "Locked Valve Log", that this event did not involve an actual mispositioning of the valve.

The valve is locked closed but is routinely opened to provide fire protection (water to the hose reels) for cutting, grinding or welding operations in the Reactor Building. Cycling of the valve is recorded in the "Locked Valve Log". Accordingly, the inspector concluded that the LER was submitted to meet the administrative reporting requirements of 10 CFR 50.73 and had no effect on the operation of the Reactor Building Fire Service System nor the health and safety of plant personnel and the public. The inspector confirmed that the necessary procedural changes were completed by reviewing the Change Notices to the controlling procedures 4210-OPS-3810.01 "Fire Protection System", and 4210-SUR-3244.01 "Containment Integrity Verification", both procedures were reviewed and approved by the NRC per the requirements of Technical Specification 6.8.2.

On August 27, and September 3, 1985, the inspector entered the 305' el. of the Reactor Building annulus area, and verified that the valve was in the required position.

-- LER 85-05, dated April 5, 1985, addressed a licensee identified condition in which the Reactor Building (RB) internal pressure was considered to be slightly positive between the hours of 1900 through 2313 on March 16, 1985. The root cause of this condition is attributed to a drop in atmospheric pressure from 30.27 inches of mercury (in. Hg) to 30.00 in. Hg while the RB was isolated and the RB Purge Exhaust System was shut down. A relieving operating shift identified this condition by interpreting the recording line of the strip chart recorder for RB Internal Pressure Indicators BS-PT-1412 and BS-PT-4388 as being slightly above zero. Due to the inherent inaccuracy of the strip chart, the interpretation was contrary to that of the previous shift. However, upon making this judgement, the licensee determined that the one (1) hour time clock of the action statement of the Technical Specifications Limiting Condition of Operation 3.6.1.4 had been exceeded and started the RB Purge Exhaust System to return the RB internal pressure to less than zero (0) psig. The negative pressure condition was achieved within thirteen (13) minutes, at 2313 hours.

The inspector reviewed plant parameters relating to the event, examined associated instrumentation, and discussed the event with Plant Operations and Instrumentation & Control personnel. The inspector determined that at the time of the event, the recording line of the strip chart recorder for the Technical Specification required instruments BS-PT-1412 and BS-PT-4388 was essentially indistinguishable from the zero (0) psig data line of the strip chart. The inspector examined data taken from non-Technical Specification instruments, the (1) RB Heise gauge that measures differential pressure between the RB and the Fuel Handling Building (FHB), and (2) the RB absolute pressure Heise gauge. The differential pressure indicated that the RB was slightly positive with respect to the FHB, but the absolute pressure gauge indicated that the RB pressure was negative (never exceeding 29.78 in. Hg) with respect to atmospheric pressure (ranging from 30.27 in. Hg to 30.00 in. Hg during the event).

Based on these instrument readings, the inspector concluded that the licensee made a conservative judgement in identifying the event. If

a slight positive pressure did occur in the RB, it did not present a sufficient driving force nor have a pathway for any unfiltered, unmonitored release. Since the RB was positive with respect to the FHB, flow would be to the FHB not the outside environment.

Additionally, during the four (4) hours of the event (on a Saturday evening), the RB was isolated, the RB purge secured, and no operations were going on inside the RB which would increase the airborne activity level. Such conditions would keep the airborne activity level to a minimum. An event of this nature is bounded by the analyses found in the NRC-approved Safety Evaluation Reports for "Removing the RB Equipment Hatch" and for "Opening Both Airlock Doors" for transferring large pieces of equipment into/from the RB. In both cases the effects of a positive internal pressure have been determined to have negligible impact on safety and health of plant personnel and the public.

-- LER 85-04, dated April 5, 1985, addressed a licensee identified condition in which on two (2) separate occasions the Reactor Building (RB) Internal Pressure Indication, BS-PT-1412, recorded a value slightly greater than zero (0) psig on March 5, 1985 and on March 26, 1985, while both Equipment Hatch Personnel Air Lock Doors were open and the RB Purge Exhaust System was operating. The cause of these events has been determined to be erroneous internal pressure indications during "double door" RB entries.

The inspector reviewed plant conditions relating to these events. Through observations of double door RB entries, the inspector determined that air flow was into the RB based on airlock streamer indications and anemometer measurements even though the readings on BS-PT-1412 were slightly greater than zero (0) psig. The inspector determined that the licensee was complying with the requirements of an NRC approved procedures 4210-OPS-3240.01 "Reactor Building Entry" by observing the licensee perform the sequence of steps outlined in the procedure for opening of both doors of airlock No. 1. The inspector observed that air flow into the RB exceeded 200 feet per minute and that the licensee made hourly anemometer measurements.

3.0 Licensee Action on Previous Inspection Findings

(Closed) Violation (320/85-01-05): Failure to check individual's available dose resulting in exceeding an administrative dose limit.

The inspector reviewed the licensee's corrective actions including disciplinary action and re-emphasis of the importance of determining each worker's available dose prior to authorizing work in the Reactor Building. In addition, the Radiation Work Permit (RWP) procedure (9200-ADM-4110.04, Rev. 1-00) was revised to explicitly require that the "Required Available Exposure" block on the RWP be checked against a worker's available dose for the calendar quarter prior to allowing entry into an RWP area. Finally, a change was made to the computerized "Rem-on-line" system such that an attempt by a control point technician to enter a worker on an RWP without that worker having adequate available dose will result in a "flag" appearing on the CRT screen. This "flag" consists of a message to the technician explicitly stating that the

"Individual does not have required available exposure for this entry..." and should be effective in helping to prevent a recurrence of this type of incident.

The inspector had no further questions.

(Open) Inspector Follow Item (320/84-04-04): Establish training program for people at planning and direction level (including contractors) who affect ALARA implementation.

Training requirements are included in a new Procedure 4000-ADM-4010.01, "TMI-2 ALARA Program", effective July 1, 1985, which implements Procedure 4000-PLN-4010.01, "TMI-2 ALARA Program Plan; issued January 31, 1985. Procedure 4000-ADM-4010.01 requires the availability of training courses within six months of the issue date of the procedure. The licensee's August 2, 1985 letter clarifies and confirms the licensee's intention to schedule training in accordance with this procedure. The licensee's implementation of these procedures will be reviewed during future inspections of the licensee's ALARA Program.

4.0 Transfer of Water from CO-T-1A to CC-T-1

Per the requirements of Technical Specification 6.8.2, the inspector reviewed Special Operation Procedure (SOP) 4215-3526-85-204 for the transfer of 115,000 gallons of contaminated water from the Condensate Storage Tank (CO-T-1A) to the EPICOR II Receiving Tank (CC-T-1). The transfer is to be made in two batches (50,000 gals. and 65,000 gals.), in preparation for processing the water through EPICOR II and eventual return of the processed water to CO-T-1A, following the desluging of CO-T-1A. Through review of licensee Chemistry records, the inspector determined that the water contained approximately 100 MPC Sr-90, 20 MPC H-3, and 0.6 MPC Cs-137. The inspector approved the SOP with the stipulation that walkdowns of the pump, valves, and transfer hose be conducted at least every two hours to identify any leakage or malfunction. The time required to transfer one batch is about 32 hours.

During the transfer of batch #1 on August 27, 1985, the inspector walked down the system to confirm that flanges were bagged, and that the bagging contained absorbent material, that valves were properly aligned, that system walkdowns by licensee personnel were completed within the specified frequency, and that the licensee complied with the requirements of the SOP. The inspector confirmed that the transfer of the first batch was completed on August 28, 1985.

No violations were identified.

5.0 Canister Storage Modules

During the reporting period, two fuel canister storage racks were received on site from the Nuclear Engineering Services Company (NES). When these racks were up-ended to the normal vertical position from the shipping dunnage and the black plastic wrapping removed, it was noted that the plastic wrapping was sticking to some of the stitch welds which had engaged the dunnage during shipment. Further examination revealed

that the welds were cracked longitudinally and the crack front had "grabbed" the plastic cover.

Of the nominal 48 welds on each rack which would have engaged the shipping dunnage, 17 showed visual signs of cracking on one fuel rack and 12 on the other. At the request of the NRC, the licensee performed Dye Penetrant Testing (PT) on 16 of the welds which engaged the dunnage on one rack and 17 similar welds on the other rack. Seven additional weld cracks were identified on the former rack and two additional cracks were identified on the latter rack.

Subsequent to the reporting period, a third fuel canister rack was received on site. PT examination of the welds which had engaged the dunnage during shipment did not reveal any cracking. A metal shim and rubber backing were used to cover the weld area which engaged the dunnage for shipment of the third rack.

A fourth rack is awaiting shipment to the site from the manufacturer.

Because of the ongoing evaluations and resolution of the stitch weld cracking, this item is considered an inspector follow up item. (320/85-16-01)

6.0 Steam Generator Sampling - Containment Double Isolation

During the review of operating procedures, Procedure No. 2104-4.132, Revision 1, dated December 1983, Sampling and Chemical Addition to OTSG "A" was reviewed and it appeared that a condition could arise which violated the provisional requirement of maintaining two valve isolation at the sample sink. Additional review revealed that Revision 1 had not received NRC approval. Revision 0, the first revision to the procedure, dated February 4, 1982, was approved by the NRC under the provisions of Technical Specification Section 6.8.2.

Each month, the licensee performs a containment integrity verification using Surveillance Procedure No. 4301.M8, Containment Integrity Verification - Mode 7. This procedure requires the Control Room operators to make a notation if a containment isolation valve is allowed open under administrative controls indicating that the governing procedure has been approved per Technical Specification 6.8.2. Containment isolation valves CA-V-8 and CA-V-4A have been open since the time of the accident and Procedure No. 2104-4.132 has been the controlling procedure allowing the containment isolation valves to be open. The operators have been notating on the data sheet for the Surveillance Procedure No. 4301.M8 that the Controlling Procedure No. 2104.-4.132 has been approved per Technical Specification 6.8.2. However, Revision 1 to Procedure No. 2104-4.132 does not appear to have had NRC approval. There are some questions as to whether there was a need for approval of Revision 1 being that Revision 0 was approved and the changes to Revision 0 may not have required NRC approval. Also, there may have been superseding procedures listed on a procedural matrix which would have indicated NRC approval of the governing procedures for controlling effluents through the open containment isolation valves. Because all the issues associated with this procedural review have not been resolved this item is considered an unresolved item. (320/85-16-02)

7.0 Water Spill During DWCS Pre-Op Test

On August 14, 1985, at 11:45 AM, a hose connection on a test manifold failed spilling an estimated 3,000 gallons of water into the Reactor Building sump. The spill occurred during a pre-operational leak test of the "A" filter train of the Defueling Water Clean-up System (DWCS) (per UWI 4370-3525-85-P433). During the test, two hose clamps failed allowing a hose to separate from the hose barb. For the purposes of the test, water was being transferred from the deep end of the fuel transfer canal (FTC) through the DWCS and returned to the FTC. As a result of the hose separation, the FTC level decreased by about 10 inches. No environmental releases or increases in Reactor Building radiation levels occurred. The FTC water level was restored to normal by transferring water from the Borated Water Storage Tank. The licensee's corrective action included removing the test hoses and flanging the test valves directly to the DWCS permanent piping.

No violations were identified.

8.0 Routine Health Physics and Environmental

a. Plant Tours

The NRC site radiation specialists performed routine plant inspection tours. An off-shift tour was made on August 14, 1985. These inspections included all radiological control points and selected radiologically controlled areas. Items inspected included:

-- Access control to radiologically controlled areas

-- Adherence to Radiation Work Permit (RWP) requirements

-- Proper use of respiratory protection equipment

-- Adherence to radiation protection procedures

-- Use of survey meters, including personnel frisking techniques

-- Cleanliness and housekeeping conditions

-- Fire protection measures.

These inspections resulted in the following:

1) Anteroom Personnel

During a tour of the plant by the Chief of the TMI-2 Project Section, a worker in the personnel hatch anteroom was observed demonstrating a lack of alertness. A similar event was noted during inspection 50-320/85-10. The individual was seated in a resting position with his eyes closed and his head resting against the wall. The inspector noted that the area was a posted contaminated area and the individual was dressed in anticontamination clothing. The individuals are posted to respond to problems in the Reactor Building as well as assist

exiting workers. This individual's condition appeared to impair his ability to fully discharge his duties. The licensee has reduced the duration of workers' anteroom tours and now requires that the workers stand. The suitability of workers' condition for performing assigned work will be reviewed during future inspections. (320/85-16-03)

2) Issuance of Respiratory Protection Equipment

During routine tours of the plant areas and control points on August 27, 1985, the inspector evaluated the condition of respiratory protection equipment. The disbursing points for respiratory protection equipment are the HP-2 control point and the Reactor Building Personnel Access Facility (PAF).

At the control point, the inspector determined that four negative pressure respirators had not been inspected for more than thirty days as required by procedure. To preclude use of the expired equipment, the inspector informed Radiological Controls Personnel of his findings. The out-of-date respirators were isolated and bagged for return to the inspection/repair facility.

Upon examination of the powered air purifying (PAPR) units stored at the PAF, the inspector determined that 14 of 30 units examined had expired inspection dates. The licensee also isolated, bagged and returned these units for inspection.

TMI Administrative Procedure 4213-ADM-4020.01, Inspection and Maintenance of Respiratory Protection Equipment, Revision 0-00, effective 6/28/85 states that respiratory protection equipment in routine use shall be inspected every thirty days. The failure to inspect respirators ready for use in accordance with the above requirements is an apparent violation. (320/85-16-04)

The inspector also identified the following inadequacies concerning respirator storage and inspection:

-- Inspections and inventories of authorized storage/issue locations were not being performed adequately.

-- Unauthorized storage locations were being used, whereby devices could be stored without being inventoried or inspected by respiratory personnel.

-- The slowdown of respiratory device use, which had been developing since mid-July, was the apparent cause of the expired devices being available for use. The inspector determined there was a possibility that individuals could have worn an expired device.

The licensee began an investigation of this incident to determine if any individual had worn an out of date respirator. The investigation had not been completed by the close of the inspection. Preliminary results indicate that 16 workers may have worn a respirator beyond the 30-day inspection period. The licensee is performing analyses to determine if an uptake of radioactive material occurred. These analyses include recalculating breathing zone air samples to determine the potential level of radioactivity inhaled and performing whole body counts on the individuals. The licensee inspected the affected respirators. No problems with respirator sealing surfaces were identified. Whole body counting to date has shown no internal deposition. The investigation also identified that respirator maintenance was not being properly recorded on equipment history cards.

The licensee has issued temporary changes to the inspection procedures to require documentation of each inspection and requiring respiratory protection personnel to perform and document inventory of devices at each of the issue points. The licensee has instructed personnel that the issue points are to be used and no unauthorized storage locations should be used in the future.

The inspector will review the completed investigation and resulting corrective actions in a future inspection. (320/85-16-05)

b. Measurement Verification

Measurements were independently made by the inspector to verify the quality of licensee performance in the areas of radioactive material shipping, radiation and contamination surveys, and onsite environmental air and water sampling analyses. Licensee data was consistent with the NRC measurements except as described in paragraph 8.0.f.

The inspector reviewed the radiological controls applied within the plant. Appropriate postings, surveys, and controls were observed during inspector tours during day-shift and off-shift hours.

No violations were identified.

c. Reactor Building Entries

The inspectors monitored RB entries conducted during the inspection period. The inspection activities included review of selected documents and direct observations of RB entries. The following items were verified on a sampling basis.

-- The RB entry was properly planned and coordinated to assure that task implementation including adequate As Low As is Reasonably Achievable (ALARA) review, personnel training, and equipment testing.

-- Radiological precautions were planned and implemented including the use of an RWP and specific work instructions.

-- Specific procedures were developed for unique tasks and were properly implemented.

Entries 668 through 688 were conducted during this inspection period.

No violations were identified.

d. NRC Reactor Building Entry

Two radiation specialists entered the Reactor Building as a team during entry number 685 on August 29, 1985, to assess radiological and industrial safety conditions. The inspectors traversed portions of the fuel transfer canal, the top of the "A" D-ring, approximately 60% of the 305' level, and approximately 75% of the 347' level of the Reactor Building outside of the D-rings. Housekeeping and industrial safety were satisfactory.

Considering the types of work ongoing in the building and restrictions of the Reactor Building, housekeeping and industrial safety appeared generally good.

Radiation levels were measured throughout the areas traversed, including the fuel transfer canal and the defueling platform mounted above the reactor pressure vessel. The results of radiation level measurements were in good agreement with levels reported by the licensee. Observations in the fuel transfer canal did not reveal any leakage from the seals at each end of the dam at the north end of the canal.

e. Equipment Decontamination Incident

(1) Background

A Radiological Awareness Report (RAR) No. 85-048 was initiated on May 6, 1985 by a Radiological Controls Technician to address concerns related to decontamination of a cover plate which had been used on a 4' x 4' cylindrical steel spent filter liner stored in the Fuel Handling Building Truck Bay (305' elevation). On May 6, 1985 near 8:30 AM, two workers began to decontaminate the filter liner cover plate located in the Kelly Enclosure temporary decontamination room in the Truck Bay. When the Radiological Controls Technician assigned to monitor the job arrived at 9:00 AM to survey another piece of material (also a cover plate) in the Kelly Enclosure, it was discovered that a survey had not been documented at the time the cover plate being decontaminated was placed in the Kelly Enclosure. The Radiological Controls Technician stopped the decontamination job, informed supervision, performed a survey, documented the survey, and identified the persons involved.

(2) Inspector Findings

The inspector, through interviews and document reviews, made the following determinations:

-- The liner cover plate had been removed from the liner on April 26, 1985, as part of the process of sealing the liner and removing the liner from the Truck Bay.

-- The Kelly Enclosure was the appropriate location to place the cover plate at the time. The cover plate had been surveyed, found to need decontamination, wrapped in Herculite plastic, and placed inside the Kelly Enclosure. A "radioactive material" tag had not been placed on the wrapped cover plate, since the technicians and supervisors involved were under the incorrect impression that the Kelly Enclosure postings of radioactive material and radiation area were sufficient and that the cover plate would be surveyed again prior to starting decontamination.

-- The discussion in the morning of May 6 between the decontamination workers and radiological controls personnel concerning the work to be performed had confused two similar items. The decontamination personnel were discussing the wrapped and contaminated cover plate while the radiological controls personnel were discussing the already decontaminated cover plate which was ready for release. As a result, the decontamination workers were working on the contaminated cover plate while the radiological controls personnel thought they were working a cover plate ready for release.

-- No personnel contamination was detected and no personnel exposure limits were approached. Breathing zone air samples taken in conjunction with work on the contaminated cover plate showed no increased levels of airborne contamination at any time. Whole body counts performed on the individuals involved indicate that no uptake of radioactive material occurred.

-- All of the groups involved: radiological controls technicians, radiological controls supervisors, decontamination personnel and their foremen, were counseled in the importance of proper surveys, documentation and notifying waste management and decontamination personnel when an item is placed in the Kelly Enclosure for decontamination.

(3) Conclusions

The licensee appropriately followed its procedures in investigating and evaluating this occurrence. No personnel uptake was detected and no exposure limits were exceeded. Radiological Controls personnel and decontamination personnel were counselled. Radiological Awareness Report 85-048 was

responded to on June 1, 1985 and the file closed on July 3, 1985.

The inspector had no further questions.

f. Radioactive Material Shipments

The NRC site radiation specialists inspected selected TMI-2 radioactive material shipments during the inspection period to verify the items listed below.

-- The licensee had compiled with approved packaging and shipping procedures.

-- The licensee had prepared shipping papers, which certified that the radioactive materials were properly classified, described, packaged, and marked for transport.

-- The licensee had applied warning labels to all packages and had placarded vehicles.

-- The licensee had controlled the radioactive contamination and dose rates below the regulatory limits.

Inspector review of this area consisted of (1) examination of shipping papers, procedures, packages, and vehicles, and (2) performance of radiation and contamination surveys of shipments on August 21, August 29, August 30, and September 4, 1985.

Following notification that radioactive waste shipment RS-85-079-II was ready to leave the site, a wooden container enclosing samples of electrical cables from the Reactor Building was surveyed by an NRC radiation specialist. It was noted that the maximum radiation level at the surface of the container was approximately three times the 8 millirem recorded on the container survey included in the shipment documentation package. An independent survey by a second NRC radiation specialist using a different survey instrument confirmed the higher reading. The results of a subsequent survey by a licensee technician was consistent with the radiation specialists' survey results. The shipment was subsequently released since the radiation levels were well within regulatory limits.

This item remains unresolved pending a review of the circumstances surrounding the original survey of the container and the manner in which records of surveys of the container and the shipment were corrected. (320/85-16-06)

g. ALARA Program - Allegation

An individual made an allegation that the licensee's ALARA program was deficient; in that, there had been misuse of Thermoluminescent Dosimeters (TLD) on Radiation Work Permits (RWP); and there was no safety representative on the backshift.

An investigation of the allegation was done by reviewing procedures, Radiation Work Permits, training records, TLD issuance and interviewing of individuals.

The inspector reviewed the following documents:

-- 9200-ADM-4010.02, Revision 1, "ALARA Review Procedure"

-- 9200-ADM-4110.04, Revision 1-00, "Radiation Work Permits"

-- 9000-ADM-4241.07, Revision 0, "Personnel Dosimetry Requirements"

-- 9000-ADM-4241.01, Revision 0, "Dosimetry Issue and Data Handling"

-- 210-ADM-2623.04, Revision 0, "General Employee Training Program, Units 1 and 2"

-- 1054.16, Version 3, "TMI-2 Emergency Plan Implementing Procedures; Contaminated Inquiries/Radiation Overexposure"

The alleger stated that TLD's were improperly issued. The inspector reviewed licensee policy and found that any employee working at the site may request one. However, to perform work in an area requiring a Radiation Work Permit (RWP), an individual must successfully complete General Employee Training, Radiation Worker Training and be issued a TLD.

Records of fifteen randomly selected employees were reviewed to determine qualifications in GET, Radiation Worker Training, and issuance of TLD. Radiation Work Permits records were reviewed to determine if any worker not qualified in Radiation Worker Training had been permitted on an RWP. No discrepancies were found.

During backshifts, i.e., evening and night shifts, the Shift Foreman is designated as the Emergency Director. Any individual injured receives several levels of treatment based on the severity of the injury and if exposed to or contaminated by radioactive materials. The Shift Foreman and the Group Radiological Control Supervisor are notified of any injury. In the case of backshift, first aid is administered by the Fire Brigade personnel. If off-site medical assistance is required, Dauphin County Emergency Operations Center is called and an ambulance is requested. If the individual is contaminated with radioactive material, the individual is transported to Milton S. Hershey Medical Center for treatment. The ambulance service is available on 24 hour call and has been prompt in responding to requests.

No violations were noted. This completes NRC review regarding these allegations.

9.0 Unresolved Items

Unresolved items are findings about which more information is needed to ascertain whether they are violations, deviations, or acceptable. Unresolved items are addressed in paragraphs 6.0 and 8.0f.

10.0 Inspector Follow Items

Inspector follow items are inspector concerns or perceived weaknesses in the licensee's conduct of operation (hardware or programmatic) that could lead to violations if left uncorrected. Inspector follow items are addressed in paragraphs 5.0, 8.0 (1), and 8.0 (2).

11.0 Exit Interview

The inspectors met periodically with licensee representatives to discuss inspection findings. On September 10, 1985, the inspectors summarized the inspection findings to the following personnel at the exit meeting:

J. Auger, Licensing Engineer
J. Byrne, Manager, Licensing, TMI-2
W. Conaway, Manager, Radwaste Support
W. Craft, Manager, Radiological Controls Field Operations
E. Gee, Deputy Manager, Radiological Controls
W. Heysek, TMI-2 Audit Supervisor
S. Levin, Director, Site Operations
B. Parfitt, Supervisor, Respiratory Protection
J. Renshaw, Manager, Radwaste Management
R. Rogan, Director, Licensing and Nuclear Safety
M. Slobodien, Manager, Radiological Engineering
F. Standerfer, Director, TMI-2

At no time during the inspection was written material provided to the licensee by the TMIPO staff except for procedure reviews pursuant to Technical Specification 6.8.2.