U. S. NUCLEAR REGULATORY COMMISSION

Report No. 50-320/85-21

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: October 7, 1985 - November 8, 1985

Inspectors: R. J. Cook 12/12/85
R. Cook, Senior Resident Inspector (TMI-2) date signed

12/13/85
T. Moslak, Resident Inspector (TMI-2) date signed

Jack M. Bell 12/12/85
J. Bell, Senior Radiation Specialist date signed

David J. Collins 12/12/85
D. Collins, Radiation Specialist date signed

Larry E. Myers 12/12/85
L. Myers, Radiation Specialist date signed

Approved By: C. Cowgill III 12/18/85
C. Cowgill, Chief, TMI-2 Project Section date signed

Inspection Summary:
Areas Inspected: Routine safety inspection by site inspectors of plant operations (long term shutdown) including review of sampling and chemical addition procedures for the "A" OTSG; radiation surveys of radioactive material shipments; misclassification of radioactive waste being sent for burial; failure to inspect respiratory protective equipment within the required period; upgrading and implementation of new extremity dose assessment methods; ensuring clarity and accuracy of strontium-90 chemical analysis procedures; defueling operator training; defueling operations; review of filter canister QA documentation; reactor building entries to observe proposed restaging of the missile shields and radiological and industrial safety conditions; plant tours to observe radiation protection control; radiation measurement verification; radiological waste management; unmonitored release pathways; exemption from 10 CFR 61.55 and internal deposition of radioactive material. The inspection involved 327 inspector hours.
Results: A violation was identified for failure to comply with a double isolation valve requirement of the OTSG "A" Sampling and Chemical Addition procedure. A second violation was identified for failure to perform adequate surveys prior to workers entry into a highly surface contaminated area.

DETAILS

1.0 Ongoing Recovery Operations

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 backshift hours were routinely conducted. The Shift Foreman's Log and selected portions of the Control Room Operator's Log were reviewed for the period October 7 through November 8, 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 Action on Previous Inspection Findings

(Closed) Unresolved Item (320/85-16-02): Sampling and Chemical addition procedures.

As stated in Inspection Report 50-320/85-16, Revision 1 to Procedure No. 2104-4.132, Sampling and Chemical Addition to OTSG "A", was not reviewed and approved by the NRC. However, the licensee, when considering whether there was a need for NRC approval for the revision, determined that the revision did not directly relate to core cooling; could not cause the magnitude of a radiological release to exceed limits established by the NRC; could not increase the likelihood of failure in systems important to nuclear safety and radioactive waste processing or storage; and could not alter the distribution or processing of significant quantities of stored radioactivity or radioactivity being released through known flow paths and therefore did not require NRC approval. In August and September, 1985, the licensee updated many of their administrative procedures and as a result, procedures which were originally approved by the NRC, now require NRC approval for all subsequent changes. Also, the licensee has done an audit of the listed procedures in Surveillance Procedure No. 4210-SUR-3244.01 which is the current successor to Surveillance Procedure

No. 4301.M8, Containment Integrity Verification and identified those revisions which were not approved by NRC even though the original versions were NRC approved. Updating of procedure cover sheets to reflect NRC approval should remove any question and/or ambiguity when the operators are considering whether the procedure has been approved by the NRC.

During the NRC review of Operating Procedure No. 2104-4.132, Revision 1, dated December 6, 1983, Sampling and Chemical Addition to OTSG "A", it was determined that a condition could arise which could violate the provision of Step 2.2.3 that "...two valve isolation is provided at sample sink." by closing only the OSTG "A" drain line isolation valve as stated in Step 4.3.8 - the last step of the procedure. This leaves valve SS-V-51, the second isolation valve, still open from prior Step 4.3.4 thus negating the provisions of Step 2.2.3 for two valve isolation.

Technical Specifications Section 6.8.1 requires that procedures shall be established, implemented and maintained which cover sample procurement. Contrary to this, Operating Procedure No. 2104-4.132, Revision 1, Sampling and Chemical Addition to OTSG "A", was written in such a way that the procedure would not accommodate Step 2.2.3 which states, "Do not close CA-V-8 or CA-V-4A, two valve isolation is provided at sample sink." Two valve isolation is not provided at the sample sink by the procedure. The failure to maintain two valve isolation at the sample sink because of conflicting procedural requirements is an apparent violation. (50-320/85-21-01)

Procedure No. 2104-4.132 has been deleted and replaced by Procedure No. 4212-OPS-3562.01, Sampling and Chemical Addition to OTSG "A" during the upgrading and redesignation of operating procedures. Revision 1 to this procedure (Procedure No. 4212-OPS-3562.01) included as the final step (Step 6.3.9) to "close SS-V-51" and thus establish two valve isolation.

(Closed) Unresolved Item (320/85-16-06): Radiation surveys of radioactive materials shipments.

The inspector reviewed the circumstances surrounding the original survey of the package in question and determined that a reasonable attempt was made to measure radiation levels at the package. As described to the inspector, the survey process was such that a small area of the package surface was obstructed and not accessible at the time of the survey. The licensee described measures taken to help ensure more thorough surveys, including counseling of those involved and discussion of the survey process at employee meetings and briefings.

In reviewing the survey record, the inspector determined that the individual surveying the package on the transporting vehicle had changed the radiation level figure on the survey form (rather than crossing it out, entering a new figure, initialling it and obtaining the foreman's initials on the record). This was done following the individual's second survey of the package because of a feeling by the individual that the survey was simply being completed rather than being changed. This assessment of the situation by the individual was based on the short time interval between the initial survey of the package on the transporting

vehicle and the second survey (requested by the inspector). The individual has been counselled and now understands that any time any radiation level figure is changed on a survey form, regardless of the elapsed time between measurements, the incorrect figure must be crossed out and the new figure recorded and initialled and, if the form has become a record (i.e. signed by the foreman), the foreman must also initial the change.

The inspector had no further questions.

(Open) Violation (320/85-15-01): Misclassification of radioactive waste being sent for burial.

The licensee has taken action to preclude recurrence of the iolation. The inspector verified licensee actions consisting of modifications to its procedures 4214-ADM-4450.01, "TMI-2 Radioactive Material Shipment Portfolio Preparation;" and 1501-ADM-4450.01, "Packaging of Radioactive Waste for Disposal at a Commercial Disposal Facility." Additionally, the licensee is proceduralizing guidance to its shipping coordinators to assure uniformity and quality in shipment preparation methods. The licensee is also taking action to assure the continued accuracy of the computer programs which are used to perform waste classifications per the requirements of the Department of Transportation in 49 CFR and the requirements of the NRC in 10 CFR Parts 20, 61, and 71.

The procedure changes incorporate a form to be used when Waste Management personnel request a curie content determination of a waste package by Radiological Engineering. The form specifically asks for verification of the assumptions to be used in making the curie determination or notification that a change was made. Additionally, a waste classification is now required on each package which does not fall within a normal waste stream category based on that package's specifically determined content. This had not been a routine practice, but is now required by the additional procedure guidance.

The inspector determined through observation and discussion that all the affected individuals had received training in the appropriate revised procedures. The licensee expects to have implemented the final procedure changes by December 31, 1985.

This item remains open pending implementation of the permanent procedure changes.

(Open) Violation (320/85-16-04): Failure to inspect respiratory protective equipment within the required period.

The inspector determined that Temporary Change Notices (TCNs) had been issued to procedures 4213-ADM-4020.01, "Inspection and Maintenance of Respiratory Protection Equipment" and 9000-ADM-4020.06, "Issue and Control of Respiratory Protection Equipment." These changes were effective September 6, 1985 and will expire on December 6, 1985. The procedure changes include instructions to personnel who inventory respirators to ensure that devices available for issue have inspection due dates far enough in the future to allow a reasonable time period for

their use and thereby help to prevent outdated equipment from being used, and to ensure that the inspections performed on respirators removed from service are properly documented. The inspector determined through observations and interviews that personnel were familiar with the temporary changes to the procedures.

The inspector has no further questions at this time. However, this item remains open pending issuance of permanent procedure changes. The licensee has stated that the permanent procedure changes are expected by December 31, 1985.

(Open) Inspector Follow Items (320/84-21-03 and -04): Resolve administrative, logistical and computer software issues, upgrade and implement new extremity dose assessment method.

The inspector determined that although significant progress has been made toward developing and implementing a revised extremity dose assessment and recording system, much remains to be done. The inspector was informed that although necessary changes to the radiation exposure management system (REMS) have been identified, reprogramming of the system and retroactive data modifications have yet to be accomplished because of higher priority work on the REMS. Consequently, the inspector has determined that the licensee's new action date of October 1, 1986 as proposed in its letter 4410-85-L-0216 dated October 31, 1985, is acceptable.

(Open) Inspector Follow Item (320/85-20-01): Ensure clarity and accuracy of strontium-90 chemistry analysis procedures.

In accordance with Confirmatory Action Letter 85-16, the licensee has reviewed and clarified the procedures used for analyses of strontium in solid samples. The licensee issued temporary change notices (TCNs) for two procedures; 4212-CHM-3013.70, "Radioactive Source Preparation" and 4212-CHM-3012.89, "Operation of the Beta Spectrometer."

The inspector verified the adequacy of the TCNs and determined that the appropriate Chemistry personnel understood the changes. The inspector noted that the liquid strontium-90 standard certification document carries a specific notation that strontium-90 was in equilibrium with yttrium-90. The inspector verified that department management has requested that the Training Department provide training of Chemistry Department personnel in the revised procedures through normal initial, cyclic and refresher training.

The inspector had no further questions, but this item will remain open pending implementation of permanent procedure changes.

3.0 Defueling Operator Training

On October 29, 1985, the inspector audited training records for those auxiliary operators who were scheduled to conduct initial defueling operations on October 30th. Upon reviewing the practical factor qualification cards, the inspector identified two cards that had not been

signed by the Task Examiner, indicating failure to complete specific tasks of Phase 1 training.

The cards indicated that both individuals had not been qualified to operate the Service and Jib Cranes and the long handled measuring tools; and that one operator was not trained on operating the End Effector Handling Tool and the End Fitting Loading Tool.

The inspector informed the Operator Training Manager of these findings. Subsequently, licensee representatives took the following corrective actions:

-- Identified that one operator had received the required training but that the Task Examiner had inadvertently failed to sign the qualification card. The card was subsequently signed by the Task Examiner.

-- Identified that the second operator had not received the training. The operator was trained on the specific items on October 30, 1985, and was determined to be qualified.

-- Assigned a Site Operations Engineer to establish a system for ongoing review of training records of operators performing defueling tasks, to insure that personnel had received the required training.

The inspector had no further questions on this matter.

4.0 Defueling Operations

On October 30, 1985, the licensee initiated preliminary defueling activities. These activities involved rearrangement of core debris within the reactor vessel to allow complete installation and uninterrupted rotation of the Canister Positioning System (CPS). Approximately 2,000 pounds of debris, consisting mainly of end fittings with attached fuel rods, were moved from the northern quandrant within the reactor vessel to the southern regions. Video inspections confirmed that an appropriate clearance had been excavated for CPS operation.

The inspector observed these activities daily and determined that the licensee conducted in-vessel operations in accordance with the provisions of the Safety Evaluation Report (SER) and procedures approved by the NRC staff.

On November 12, 1985, subsequent to receiving NRC approval, the licensee began "Early Defueling" activities. These activities are limited to the picking and placing of core debris into canisters, the removal of these filled canisters from the reactor vessel, and their transfer to and storage in the "A" Spent Fuel Pool. Using a long handled vise grip tool, the defueling operators placed a broken piece of fuel rod into a canister. On November 14, 1985, operators picked and placed the first piece of an end fitting into a canister. Difficulties were encountered with end fittings which are fused together and single assemblies that had

rods randomly oriented. These assemblies must be reduced in size in order to fit through the canister seal cover.

The inspector observed these activities on a daily basis to assess compliance with applicable requirements. No violations were identified.

However, the inspector noted several weaknesses, e.g. garbled communications, that detracted from the overall efficiency of the defueling operations. Licensee performance in conducting defueling operations will continue to be monitored in future NRC inspections. (320/85-19-02)

5.0 Review of Filter Canister Documentation

The inspector reviewed licensee documentation for acceptance of filter canisters serial numbers F401, 402, 403, and 404 for use. This documentation consisted of the quality assurance documentation package that accompanied the canisters from the manufacturer and a special check list performed by Bechtel North American Power Corporation. The inspector reviewed selected certificates of compliance, Certified Material Test Reports (CMTR), applicable drawings and selected Nuclear Engineering Services Manufacturing material assembly travelers. The inspector confirmed that material for the canister upper head material, and canister plugs conformed to design.

During this review the inspector identified that the CMTR provided by the manufacturer for the canister bottom end plug indicated that the plug was 304L stainless steel and not 316L as required. The inspector asked about the discrepancy. The licensee provided information that the CMTR had been included in error and showed the inspector the correct CMTR. The inspector had no further questions on this item.

The inspector noted that the canister head material could not be traced to a specific CMTR due to the method of accounting used by the manufacturer. The inspector reviewed one assembly traveler and confirmed that the material was traceable to a group of CMTR's that were of the proper material. The inspector had no further questions regarding this matter.

The inspector reviewed quality checks for poison (B_4C) manufacture and loading in tubes. These checks included selective destructive testing during manufacture, dimensional and visual checks prior to shipment for loading and quality control hold point witnessing of pellet loading by both the manufacturer and a quality representative from the licensee's Architect Engineer. In addition, the licensee stated that radiographs were performed to confirm the presence of the poison. The inspector performed independent calculations to confirm that the pellets contained the required boron ten (B_{10}) density. No unacceptable conditions were identified.

6.0 Reactor Building Entry

On November 7, 1985 the resident inspector and a staff member from the Three Mile Island Program Office (TMIPO) entered the reactor building

(RB) to examine the structural steel across the "A" D-ring and the proposed lift paths for adding additional structural steel to support the missile shields which are presently stored on top of the "B" D-ring. The licensee has proposed placing W30 x 173 structural beams on the east and west periphery walls of the "A" D-ring prior to lifting them in place over the existing structures. The W30 x 173 beams are then attached by slings to the existing W24 x 100 shapes prior to putting the W30 x 173 beams in place across the W24 x 100 beams. With the W30 x 173 beams in place the missile shields can be relocated from the "B" D-ring to the "A" D-ring. The W30 x 173 beams proposed for use on the "A" D-ring are presently used to support the missile shields on the "B" D-ring. To relocate the structural shapes from the "B" D-ring to the "A" D-ring requires some repositioning of the missile shields on the "B" D-ring. Of particular interest to the NRC inspectors was the possibility of a dropped structural shape being able to follow a path such that an incore instrument tube or any other non-isolable locations could be penetrated by the dropped "projectile". No such feasible path could be determined or established by the NRC during the physical examination of the routes and structures involved.

While in the RB the inspectors examined the portions of the fuel canal liner plate located at the north end of the pool just below the 347' elevation which had been cut away to accommodate the fuel transfer bridge mast when it is located in line and over the fuel transfer mechanism. The concrete at the nominal 2' x 2' cut away areas did not seem to have been damaged or significantly weakened by the cutting. The edges of the liner plate were sealed to the exposed concrete using an RTV type compound.

No violations were identified.

7.0 Health Physics and Environmental Review

a. Plant Tours

The NRC site radiation specialists performed plant inspection tours including all radiological control points and selected radiologically controlled areas. Backshift tours were conducted on October 20, 24, and 30, 1985. Items inspected included:

-- Access control to radiologically controlled areas

-- Adherence to Radiation Work Permit (RWP) requirements

-- Proper use and storage of respiratory protection equipment

-- Adherence to radiation protection procedures

-- Use of survey meters and radiological instruments

-- Cleanliness and housekeeping

-- Fire protection.

The inspector reviewed the application of radiological controls within the plant, the laundry facility and the Interim Solid Waste Staging Facility. The inspector reviewed the Radiological Controls Department logbooks for the period October 7 - November 8, 1985. Notations in the logbooks were appropriate to the conditions, showed attention to detail, and were properly made. The logbooks were initialed to indicate frequent review by departmental management.

No violations were identified.

b. Measurement Verification

Measurements were independently made by the inspector using NRC radiological equipment. These measurements were made to verify the quality of licensee performance in the areas of radioactive material shipping, radiation and contamination surveys, and onsite environmental air and water analyses.

No violations were identified.

c. Reactor Building Entries

A radiation specialist entered the reactor building during entry number 733 on November 7, 1985, to assess radiological and industrial safety conditions. The inspector traversed approximately 40% of the 305' level and approximately 20% of the 347' level, as necessary to reach the defueling platform located atop the open reactor vessel (RV).

Radiation level measurements made in the presence of the inspector established that radiation levels on the platform and over the opening in the platform ("slot") through which the fuel handling tools are manipulated, were as previously reported by the licensee. Measurements over the slot and on the platform side of the shielding "fence" at the edge of the slot established that the shielding was quite effective in controlling radiation dose to workers' bodies from the waist down. The ratio of the radiation level over the slot to that on the workers' side of the shield fence was about ten to one. Based on observed work practices and the higher radiation levels over the slot, it appears that the most highly exposed portions of workers' bodies would be their hands and upper bodies. All platform workers are required to wear special dosimetry sets including head, chest, right thigh, wrist and ankle TLDs. The highest reading of the head, chest and right thigh TLDs is recorded as the "whole body" dose.

The observed defueling work group consisted of four people: one video camera operator, one tool operator, one assistant tool operator, and one video console operator. All but the video console operator were on the rotatable platform. A radiological controls technician was also present on the platform to monitor radiological conditions and perform radiation surveys, e.g., surveys on tools removed from the RV. All workers appeared to function smoothly as part of the overall work group. However, the inspector did observe

some conditions that raised questions concerning radiation and industrial safety.

There was an excessive number of partial rolls of tape cluttering the platform work surface and an area monitor was not functioning properly. Immediate action was taken by the licensee to have the clutter removed from the platform. The condition of the platform surface will be observed during future inspections. With respect to the area monitor, it was later determined that the instrument had been improperly adjusted prior to being placed in the reactor building. It should be noted that there was a second area monitor at the platform slot that was functioning properly. This item will remain unresolved pending a determination of the circumstances associated with adjustment of the instrument. (50/320-85-21-03)

d. Radiological Waste Management

On August 12, 1985, Technical Specification Change, Revision 11 to 6.2.2 "Organization Plan TMI-2" was approved. A position of Manager, Waste Management was created who reports directly to Site Operations Director. The inspector discussed the changes with the licensee representatives. Administration Procedure 4210-ADM-1000.01 "TMI Unit 2 Plant Operations Organization, Responsibility and Authority" is being changed to reflect the recent Technical Specification revision. The inspector will review the procedure and organization to ensure that the Technical Specification revision has been properly implemented. (50-320/85-21-04)

e. 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 complied 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 October 9, 11, 21, 25, 29, and 30, and on November 1, 1985.

f. Unmonitored Release Pathway

The licensee identified a potential unmonitored release pathway from Contaminated Drain Tanks to the atmosphere which existed during and after the accident in March 1979.

The Contaminated Drain Tanks, WDL-T11 A and B, collect contaminated waste, i.e., contaminated with detergents and chemicals, of low level activity from decontamination showers, soiled laundry, the hot instrument repair shop, hot laboratory sinks and other miscellaneous low level sources. The expected activity of collected liquid waste in the tanks was less than 1.0 E-8 microcuries per cubic centimeter (μCi/cc) according to the Final Safety Analysis Report (FSAR). Currently, the tanks contain waste which has an activity of 1.0 E-4 to 1.0 E-5 μCi/cc.

The tanks have a local vent into the tank space of the Service Building basement (281' elevation). The collection lines have three atmospheric vents to conform to the plumbing code as required for sinks, showers and drains isolated by loop seals. The unmonitored release pathway is by the tank vents in the tank room and to the three atmospheric vents on the Service and Control Building roof. The system by design has small potential for release.

The licensee is performing evaluations to determine the amount of radioactivity released. Preliminary evaluations indicate that during and after the accident the total release has been about 1 to 2 mCi of radioiodine.

The licensee will update the Semiannual Radiological Effluent Release Report when the evaluation is complete. The atmospheric vents will be capped and a vent provided to the ventilation system of the Service and Control Building which is monitored. In addition, the licensee is evaluating other systems and tanks for unmonitored release pathways. The above will be reviewed in a subsequent inspection. (50-320/85-21-05)

g. Approval of Exemption from 10 CFR 61.55

On October 24, 1985, NRC issued an exemption to the licensee from a portion of the requirements for waste classification required by 10 CFR 61.55. The exemption allows the licensee to ship EPICOR-II spent resin liners containing up to 1.0 curie per cubic meter (Ci/m^3) of strontium-90 as Class A waste. However, all other requirements of 10 CFR 61.55, including the unity fraction limitation must be met. The State of Washington (an Agreement state) issued an amendment to the burial site license on July 17, 1985 which gave approval to bury these wastes. The inspector has reviewed shipments made under the exemption and determined the licensee has been in compliance with the exemption requirements. The first shipment was made on October 28, 1985.

The inspector had no further questions.

h. Internal Deposition of Radioactive Material

On June 14, 1985, two workers were contaminated while performing leak tests on Defueling Water Cleanup System (DWCS) valves located on the 305' elevation of the annulus in the Fuel Handling Building. After evaluating the incident, the licensee assigned an intake of 40 MPC-hours to one worker.

Description of the Incident

On June 13, 1985, a crafts foreman and a worker (Worker 1) were to enter the 305' elevation of the annulus area of the Fuel Handling Building to perform leak tests on DWCS valves located in the northwest corner of the annulus. Prior to the entry to do the work, the foreman and the Radiological Control Technician (RadConTech 1) responsible for the area discussed the scope of the work and both entered the area to establish the locations of the valves which were to be leak tested. Due to a failure in communications between the foreman and the RadConTech 1, overhead valves were surveyed but valves located approximately 15 feet from the entrance platform and accessible only by crawling over, under and around piping were not surveyed. The leak test was cancelled and rescheduled for the following day. The original foreman was unable to make the entry on June 14 and was replaced by a worker (Worker 2). A different RadConTech (RadConTech 2) was assigned responsibility for coverage of the job on June 14. The two workers stated that they believed that the foreman had adequately explained the work scope to the RadConTech 1. RadConTech 2 conferred with the workers and was told that the work was "the same as scheduled for yesterday." RadConTech 2 was briefed by RadConTech 1 concerning the scope of the work but did not discuss the work in detail with him. Based on the briefing, RadConTech 2 surveyed the overhead valves but did not survey the route to the other valves.

Upon entry to perform the leak test, Worker 1 climbed a ladder to the overhead valves. Worker 2 traversed the unsurveyed route to the valves 15 feet from the platform by crawling over, under and around piping creating local airborne contamination from surfaces of high surface contamination. The workers were in the area for about ten minutes. After completion of the leak test the two workers exited the area and were found to be contaminated. Worker 1 was contaminated on his moustache. Worker 2 was contaminated on the chest, arms, and in his outer nasal passages (evidenced by nasal blow). Whole body counting was performed three days later on June 17, 1985. Worker 2 was found to have an uptake of Cs-137 and Sr/Y-90 attributed to this event.

Findings

The inspector reviewed Incident/Event Report No. 88-066, RWP No. 11901, the evaluations of the intake by Radiological Engineering, surveys, results, and applicable procedures. Based on this review and discussions with licensee's personnel, it was determined that:

Surveys of the area documented on the Radiation Work Permit (RWP) indicated contamination levels of 10,000 DPM/100 cm^2 in the general area and 480,000 DPM/100 cm^2 maximum in the area of the annulus. Airborne activity was 1.1 E-10 on June 3; the last air sample taken. Radiation levels in the area were 5 R/hr highest and 35 mR/hr in the general area. No air sample was taken immediately before or during the entries as a basis for determining the extent of airborne hazards or the need for protective equipment. The workers were not equipped with breathing zone air samplers nor was respiratory protection required.

10 CFR 20.201 requires that licensees perform surveys appropriate to the circumstances. Additionally, the GPU Nuclear "Radiation Protection Plan - Unit 2," 1000-PLN-4010.01 requires that continuous air sampling be performed to supplement periodic measurements during work which has the potential for the generation of significant airborne radioactive material. Procedure 9000-ADM-4020.01 "Description and Selection of Respiratory Protection Equipment" implements the air sampling requirements. The failure to perform appropriate surveys and continuously monitor airborne activity during the leak tests is an apparent violation. (50-320/85-21-06)

During the review the inspector noted there was a failure to implement a provision of the Radiological Review, No. 50024, for "Installation of DWCS Piping and Penetration Mods in AFHB," which stated that an addendum will be added to the review for work that takes place in areas greater than 25 mR/hr. The general radiation level documented on the RWP was 35 mR/hr and no addendum was done. (50-320/85-21-07)

There was a lack of prompt and thorough follow-up of the worker's nasal contamination. The worker's whole body count was not performed until three days after the event. The "Radiation Protection Plan - Unit 2," 1000-ADM-4010.01, in Article 6 states, in part, that "internal radioactivity shall be measured promptly in each person who is suspected of inhalation of sufficient radioactivity to cause measurable internal radioactivity." The inspector expressed concern regarding the lack of appropriate guidance concerning the timeliness of whole body counts and will review procedure adequacy in a future inspection. (50-320/85-21-08)

During the review of the incident the inspector noted that several entries into the affected area were required to gather information for appropriate licensee evaluation of the incident. This indicates a potential weakness in Radiological Control Field Operations personnel training. The inspector will review this area in a future inspection. (50-320/85-21-09)

8.0 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 2.0 and 7.0.

9.0 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 2.0, 4.0 and 7.0.

10.0 Exit Interview

The inspectors met periodically with licensee representatives to discuss inspection findings. On November 15, 1985, the inspector summarized the inspection findings to the following personnel at the exit meeting:

J. Auger, Licensing Engineer
H. Behling, Manager, Radiation Health
J. Byrne, Manager, TMI-2 Licensing
E. Gee, Department Manager, Radiation Control Field Operations
S. Levin, Site Operations Director
M. Press, QA Lead Auditor
G. Skillman, Manager of Defueling

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