U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-320/89-08

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 18, 1989 - October 20, 1989

Inspectors: F. Young, Senior Resident Inspector
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Approved by: C Cowgill III Nov 29, 1989
C. Cowgill, Chief Date
Reactor Projects Section 4B
Division of Reactor Projects

Inspection Summary:

Areas Inspected: Routine safety inspections were conducted by site inspectors of defueling and decontamination activities, including the proper implementation of radiological controls, housekeeping measures, and licensee response to radiological and industrial safety events.

Results: Two personnel hazard events occurred during this period. One event involved the inadvertent handling of a piece of core debris, resulting in an over exposure to the extremities of two individuals. This event was addressed in detail in NRC inspection report 50-320/89-09. The second event involved the electrical arcing of an energized cable in the reactor building. Poor housekeeping practices were noted in the reactor building during a tour conducted by the inspectors. Poor housekeeping was concluded to be a contributing cause to both events noted in Section 2 of this report. Two examples of personnel inattentiveness-to-duty were also identified.

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DETAILS

1.0 Overview

1.1 Licensee Activities

During this report period, defueling crews started to remove the baffle plates, in the core basket assembly. This facilitated defueling behind the baffle plates in the core former region. To date, seven of eight baffle plates have been removed. Defueling crews encountered some difficulties in removing a broken portion of baffle plate 7, located in the southeast quadrant. The difficulty was due to a mass of material adhering to the back of the plate. This portion of the baffle plate is adjacent to a large hole in the baffle plate structure caused by molten debris during the accident. Upon completion of baffle plate removal and core former region defueling, the defueling will continue in the lower core support assembly region.

Defueling is scheduled to be completed by late November. Following completion of defueling, a number of metallurgical samples will be taken of lower head and incore instrumentation nozzles. These samples will be analyzed to assess the effects the accident had on the reactor vessel.

1.2 NRC Staff Activities

The purpose of this inspection was to assess licensee activities during defueling and decontamination activities. This assessment was made through observations of licensee activities, routine tours of the spaces, interviews with licensee personnel, and review of applicable documents.

The inspectors reviewed licensee's procedures implementing control on several interfacing systems with the reactor vessel to assure adequate controls were in place to prevent uncontrolled boron dilution. The inspectors also reviewed instrument calibration, switching and tagging of valves, approval authority, and responsibilities of the operations personnel.

NRC staff inspections use the acceptance criteria and guidance of NRC Inspection Procedures (NIP's). These NIP's were annotated in the Table of Contents to this report.

1.3 Persons Contacted

During this inspection, the following key licensee personnel provided information in the development of the inspectors' findings.

- -- *J. Byrne, Manager, TMI-2 Licensing
- -- P. Carmel, Waste Management Manager
- -- W. Conaway, Radwaste Support Manager
- -- L. Edwards, Quality Assurance (QA) Auditor
- -- D. Ethridge, Radiological Engineering Manager
- -- C. Incorvati, Audit Manager
- -- E. Juteau, Radioactive Material Coordinator
- -- *G. Kuehn, Site Operations Director, TMI-2
- -- *S. Levin, Defueling Operations Director
- -- *T. Murphy, Environmental and Rad Support Director
- -- W. Marshall, Manager, Plant Operations, TMI-2
- -- *C. Pollard, Manager, Rad Con Field Operations
- -- *M. Roche, Director, TMI-2
- -- *R. Rogan, Director, Licensing & Nuclear Safety, TMI-2
- -- *E. Schrull, TMI-2 Licensing Engineer
- -- R. Sieglitz, Manager, Waste Management
- -- J. Thomas, Engineer, TMI-2
- -- R. Wells, Licensing Engineer

*Attended the final management meeting.

2.0 Licensee Events

2.1 Personnel Overexposure

On Monday, September 25, 1989, two workers received a radiation exposure to the extremities (hands and wrist) that may have exceeded the quarterly dose limit of 18.75 Rem as specified in 10 CFR 20. The workers were in the flushing section of the decontamination facility (DF) in the Unit 2 reactor building. The workers were decontaminating the flushing side of the DF (including the walls, floors, gratings and various equipment in the area) using a hot water/steam flushing system. The area had previously been used on Friday, September 22, 1989 to repair a highly contaminated pump. During this evolution the workers inadvertently handled a piece of core debris, mistaking it for a bolt used in the repair of the pump.

One worker asked a Radiological Controls Technician in the Reactor Building to survey the piece in question. Radiation measurements of the material were 24 R/hr (gamma) on contact and 100 Rad/hr (beta). Based on these readings, it was determined the material was core debris. The core debris was subsequently transferred to the reactor vessel. The licensee held an event critique on September 26, 1989 and notified the NRC resident office on September 27, 1989. The licensee is performing comprehensive dose assessments for the individuals involved. The event is discussed in detail in NRC special inspection report 50-320/89-09.

Following the event, the NRC resident staff evaluated the effectiveness of the licensee's management controls for activities performed in the Decontamination Facility (DF). The staff conducted this assessment through discussions with licensee representatives, reviews of relevant documentation, and examination of material conditions in the reactor building and DF. From this evaluation, the inspectors determined the following:

--The overall housekeeping and radiological conditions of the section of the DF had significantly deteriorated prior to the event. Prior to the event the survey of the DF, determined that surface contamination levels, general area and localized radiation (gamma and beta) dose rates, and airborne concentrations had significantly increased. In light of the condition of the DF, on Friday, September 22, 1989, on-going work was stopped and the facility was posted "Keep Out. No Entry Without Rad Con Permission". One of the individuals involved in the incident stated that the housekeeping within the DF was seriously degraded in that trash and shielding materials had accumulated in the DF. Following the event, inspections of the facility performed by the NRC staff confirmed the overall poor housekeeping.

These degraded housekeeping conditions of the DF made it difficult for an adequate pre-job radiation survey to be performed to identify the true radiological conditions of the work area prior to beginning the job. These degraded conditions indicate that prior to the event that occurred on September 25, 1989, housekeeping was not aggressively controlled for this area nor had adequate pre-job planning been performed for the decontamination activities.

-While using the hot water/steam cleaning device in the DF, the workers' vision was seriously restricted. Visibility was sometimes lost as a result of the room filling with a water fog and additionally, as a result of condensation forming on the respirator face piece. Restricted visibility of the workers is considered a factor that contributed to the workers unknowingly picking up a piece of core debris. Such conditions may have been precluded, had the licensee conducted a more in-depth review in the initial equipment design review, human factor engineering evaluation for the flushing equipment, and the original industrial safety evaluation of the actual working conditions.

Based on the information gathered in this evaluation, the inspector concluded that weaknesses in management controls were a contributing factor to the personnel overexposures. Specific weaknesses include a failure to maintain adequate housecleaning in the DF, and failure to evaluate, from an industrial safety/human factor aspect, use of the hot water/steam cleaning device in the DF. A Enforcement Conference scheduled for November 17, 1989 will include discussions of these findings.

2.2 Exposure of Live Electrical Wire Discovered in Reactor Building

On October 18, 1989, an exposed live electrical cable was discovered in the Unit 2 reactor building. A chemistry technician, while attempting to sample the fuel transfer canal (FTC), released a safety chain on a catwalk leading to the FTC on the 329' elevation in the Reactor Building. The safety chain came in contact with an exposed live electrical cable on the catwalk. The resulting grounding of the cable tripped the power supply breaker. Workers in the general area of the cable, as well as workers on the north platform, were cleared from the area. The licensee dispatched two electricians to investigate and verify that power was secured.

The cable is a 480v power lead used for the plasma arc cutting equipment (PCI). Power to this cable is supplied through a four way selector switch located in a electrical panel on the 347' level of the reactor building. The switch had been danger tagged in the cavitating/pulsating water jet position. However, the electricians found the switch in the PCI position. The panel is located next to a radiological controls step off pad, and it is postulated that an individual inadvertently changed the switch position while leaning against the panel. The cable leads were found exposed. The worker who originally disconnected the lead from the PCI thought the cable was to be disconnected from the panel and stored.

The switch was restored to its required position; a field modification was processed and the cable leads to PCI equipment were disconnected from the panel and stored.

The inspector attended a critique on October 19, 1989. The licensee is performing the following action items:

-- Review reactor building for electrical cables no longer needed;
-- Investigating the need for a physical movement deterrent on the switch;
-- Having the supervisors stress the importance of good work practices and attention to detail to their workers.

The inspector concluded this event occurred due to:

-- Poor housekeeping practices (failure to remove unneeded equipment from the area);
-- Poor laborer work practices (not taping the end of the leads);
-- Poor communications between workers and supervisor (worker felt the cable was to be removed);
-- Poor supervisor follow-up and attention to detail.

Because licensee corrective actions are not complete the inspector considers this issue unresolved. This item will be followed up in a subsequent inspection report. (50-320/89-08-01).

2.3 Personnel Inattentiveness

During the inspection period, the inspector identified two instances in which licensee personnel were inattentive to their assigned duties.

The first event occurred at approximately 10:00 am on Saturday, September 23, 1989. The inspector was performing a backshift inspection of the Defueling Water Clean-up System (DWCS) operation on the 347'elevation of the Fuel Handling Building. The inspector observed an operator in the DWCS Operations Office reading unauthorized material. The inspector subsequently informed the Manager, Plant Operations, of this observation on Monday September 25, 1989. The Manager, Plant Operations, took expeditious actions to counsel the operator, to conduct an inspection of office spaces located in the protected area to locate any additional unauthorized reading material, and to inform supervisors that such practices were not to be tolerated in the work areas. Upon being informed of this incident, licensee's senior site management concurred with the actions of the Manager, Plant Operations and reiterated the policy to the site staff.

The second observation was made on Wednesday October 11, 1989 while the inspectors were touring the reactor building. While on top of the "B""D-ring", the inspectors observed a polar crane operator reclining on the walkway with his feet draped over the handrail, eyes closed, and head nodding. The Radiological Controls technician, accompanying the inspection party, roused the operator. Because of ALARA consideration, the technician also directed the individual to move to a lower radiation area. Licensee management was subsequently informed of this incident and took quick actions to counsel the individual.

These incidents and the degraded housekeeping conditions in the reactor building are considered indicative of a lapse in strict managerial controls at TMI-2.

3.0 Indications In Reactor Vessel Lower Head

During camera aided visual inspections of the Unit 2 reactor vessel lower head region on July 7, 1989, the licensee discovered crack-like indications in the proximity of two in-core detectors. The cracks appeared to be in the stainless steel cladding lining of the carbon steel reactor vessel. The cracks appeared to be approximately three to six inches in length and propagated linearly. The cracks were located near the G6 and E7 in-core instrumentation nozzles.

On August 26 and 27, the licensee inspected the crack indications and nozzle area with a color camera and several measuring and probing tools. The crack at E7 nozzles was determined to be six inches long with a crack width slightly in excess of 3/32 inches and a crack depth of 2/32 inches. The other three cracks varied from 3 to 5 inches with a width less than 3/32 inches and a depth of less than 2/32 inches. The visual inspection with the color camera showed some rust color staining in an area adjacent to the cracks. This suggests that the cracks may have penetrated the stainless steel cladding exposing the carbon steel reactor vessel surface to water. Based on crack depth information, the reactor vessel integrity was not compromised.

Preliminary analysis postulates "hot tears" in the cladding. During the accident the molten core material came in contact with the cladding material in the lower head region, raising the temperature of the vessel cladding near its melting point. This resulted in a phase change of the cladding material; and coupled with the thermal stresses involved when the molten mass cooled resulted in a grain boundary separation, produced the hot tears.

To better understand the extent and cause of this failure, this area will be included as part of the NRC lower head sample program. This will likely include metallurgical laboratory analysis of the cladding and base metal.

The inspector reviewed the visual (TV) inspection and the licensee's task force report. From this review, the inspector determined that the cracks pose no adverse impact on future defueling operations. It was also determined that the probability of driving a nozzle through the head, as previously calculated was small. However, a task force assigned to evaluate the problem recommended exercising extra caution when working around nozzles due to the potential for a leak developing. The inspector considered this to be a prudent recommendation. The licensee concluded that the change in the structural integrity of the carbon steel lower vessel head, due to corrosion, is minimal. This condition is bounded by a previous licensee safety evaluation and NRC review and does not appear to constitute an un-reviewed safety question.

As a result of the inspector's review of the visual inspection and the licensee's task force report and the inspector had no safety concerns.

4.0 <u>Accident Generated Water Evaporator</u>

The TMI-2 accident and subsequent defueling and clean-up operations used approximately 2.3 million gallons of water that require disposal. The licensee proposed to the NRC that the method of disposal would be an evaporator which would discharge a purified distillate to the atmosphere and collect the concentrate and process it as a low level solid waste. The licensee's analysis was based on a representative (base case) water sample to be processed by the evaporator. The analysis also assumed that volatile isotopes such as Tritium, and 0.1% of the nonvolatile components would be released to the environment. Based on these criteria, the evaporator must produce a minimum decontamination factor of 1000 to meet the evaluation assumptions. The NRC reviewed the analysis and documented its results in the Programmatic Environmental impact study (PEIS) Supplement 2 (NUREG-0683 Supplement 2). The PEIS concluded that if evaporator operations remain within the bounds of the assumptions and conditions stated, it would result in an acceptable level of environmental impact. The evaporation of the 2.3 million gallons will take approximately two years.

The evaporator system consists of two major subsystems. These are: (1) the evaporator and; (2) the waste handling packaging sub-system. The evaporator subsystem consists of the following components:

-- A main evaporator unit vapor re-compression unit that will distill the processed water (AGW);
-- an auxiliary evaporator that will further concentrate the bottoms of the main evaporator and provide a steam source for start up of the main evaporator;
-- and a vaporizer unit that will heat and vaporize the distillate from the evaporation process and release the vapor to the atmosphere.

The evaporator is used to distill water. The process fluid enters the evaporator heat exchanger and the fluid is heated to produce a vapor. The solids or wastes (boron, radionuclides) are basically nonvolatile and are not carried over in the vapor.

The evaporator has two modes of operation. The first is the "coupled" mode in which the evaporator and the vaporizer are operated in series. The purified distillate is the source for the vaporizer and is discharged through the process stack to the atmosphere. This is the normal mode of operation and had requirements placed upon the influent entering the evaporator (must meet "base case" conditions.)

The second is the "decoupled" mode in which the evaporator does not directly feed the vaporizer unit. This allows the evaporator to pre-process the fluid to a storage tank, to be recirculated, sampled, and analyzed prior to releasing it to the atmosphere through the vaporizer unit. The restrictions placed on the influent for this mode of operation is to ensure the solid waste (evaporator concentrates) will meet the LSA class A requirements for solid waste disposal.

The waste handling and packaging subsystem is comprised of the following:

-- a waste dryer that will further remove water from the evaporator concentrate to produce a dry solid;
-- and a packaging system that will package the dry solid waste to a form that is suitable for burial in a commercial low level radioactive waste disposal site.

Prior to delivery to the TMI-2 site, the evaporator and vaporizer unit underwent a series of operational tests at the manufacturers' (LICON) facility in Pensicola, Florida. The purpose of the tests were to define the operating characteristics and to ensure the decontamination factor (DF) of 1,000 could be achieved. The tests were run using surrogate solutions (sodium and boron added) that would simulate actual chemical composition of the water to be processed with the exception that the water contained no radionuclides.

During this inspection period, the evaporator components arrived on-site. The unit was installed on a curbed concrete slab designed to contain leakage in the event of a component failure. The licensee constructed a building to house the equipment and provide an office space for the operators. The office will serve as a radiological control point for entry into the evaporator building. The licensee is presently conducting pre-operational system checks to verify the component operability of alarms and automatic control and protection functions.

The office of Nuclear Reactor Regulation (NRR) issued a Safety Evaluation dated September 11, 1989 on the evaporation process. The following issues were evaluated.

1. Pre-processing of water to achieve "base case" radionuclide concentration;
2. Ability of evaporator to deliver a DF of 1,000;
3. The ability of the licensee to monitor effluent from process stacks and building ventilation during routine and off-normal conditions;
4. Potential accidents associated with use of the evaporator;
5. Potential for any safety problems in transporting of the evaporator concentrates to the low level water disposal site.

Based on available preprocessing systems, evaporator performance testing and the licensee technical evaluation, the NRC granted approval for the licensee to operate the system.

Prior to processing the AGW, the licensee will run a test program using surrogate solutions as the process fluid. The licensee will run three 10,000 gallon batches of a 3500 ppm boron solutions with a variable amount of sodium hydroxide being added. These tests will ensure the evaporator and vaporizer will function as designed and under the operational constraints placed upon the unit by the NRC and Commonwealth of Pennsylvania.

The inspector reviewed the licensee Technical Evaluation Report, and pre-delivery tests results and has no safety concerns. Review of the results of the surrogate testing will be documented in subsequent inspection reports.

5.0 Licensee Event Report (LER)

The inspector reviewed the LER listed below, which was submitted to the NRC Region I office pursuant to 10 CFR 50.73. Based on a review of the LER, the inspector determined the corrective actions discussed in the report were appropriate and that there were no generic issues. In addition to the technical adequacy of the LER, the compliance with the requirements of the 10 CFR 50.73 was reviewed. There were no deficiencies noted.

--LER 89-04, dated August 24, 1989, for event on July 25, 1989, "Failure of a 4160/480v Transformer". This event was reviewed in detail in Inspection Report 50-320/89-06. The inspector reviewed the LER and identified no unacceptable conditions.

6.0 Management Meeting

6.1 Regional Meeting, September 13, 1989

The NRC met with licensee management at the USNRC Region I Headquarters, King of Prussia, PA on September 13, 1989. The licensee presented, to the NRC, reports on defueling status, the accident generated water evaporation system, reactor lower head sampling program, and general topics of NRC interest. The list of attendees, agenda and slides are included in Attachment 1.

6.2 Site Management Meetings

The inspector discussed the inspection scope and findings with licensee management periodically during the course of the inspection and at a final meeting conducted October 20, 1989. Licensee management attending the exit meeting are note in paragraph 1.3.

The inspection results, as discussed at the meeting, are summarized in the cover page of the inspection report. Licensee representatives indicated that none of the subjects discussed contained proprietary or safeguards information.