U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-219/86-02

Docket No. 50-219

License No. DPR-16 Priority ______ Category C

Licensee: GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054

Facility Name: Oyster Creek Nuclear Generating Station

Inspection At: Forked River, New Jersey

Inspection Conducted: January 6 - February 2, 1986

Participating Inspectors:

W. H. Bateman, Senior Resident Inspector
J. F. Wechselberger, Resident Inspector
R. L. Nimitz, Sr. Radiation Specialist

Approved by:

ARBlough
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Reactor Projects Section 1A

2-25-86
Date

Inspection Summary:

Routine and special onsite inspections were conducted by the resident inspectors and one region based inspector (215 hours) of activities in progress including plant operations, physical security, radiation control, housekeeping, fire protection, spent fuel pool repair, and receipt, handling, and storage of new fuel. The inspectors also met with various members of management to discuss recent events and changes, followed up on concerns that arose from the vibroflotation activities associated with the proposed ESSF structure, attended a HVAC briefing, and continued a review of the Intermediate Range 10 modification.

Results:

One violation of Technical Specification (Tech Spec) requirements was identified involving hydraulic snubber operability (discussed in paragraph 6). Two unresolved items were identified involving concerns with the Intermediate Range 10 modification completed during the cycle 10 R outage. One inspector follow-up item was identified involving radcon concerns related to the dive into the spent fuel pool.

Few significant plant problems occurred during this report period and the plant continued operation near full power. Problems persisted with airborne contamination in the Augmented Offgas (AOG) building due to leaks from the AOG system; 9 people were slightly contaminated during one event when the 'B' recombiner head was removed. Licensee efforts to correct AOG problems continued. A spent fuel pool leak was located by vacuum box leak testing with helium and repaired by a diver without incident. The HVAC briefing was informative as to the progress made in restoring major HVAC systems to a functional status.

DETAILS

1. Plant Operation Review

1.1 Routine tours of the control room were conducted by the inspectors during which time the following documents were reviewed:

-- Control Room and Group Shift Supervisor's Logs;

-- Technical Specification Log;

-- Control Room and Shift Supervisor's Turnover Check Lists;

-- Reactor Building and Turbine Building Tour Sheets;

-- Equipment Control Logs;

-- Standing Orders; and,

-- Operational Memos and Directives.

The reviews indicated that the logs were generally complete.

1.2 Routine tours of the facility were conducted by the inspectors to make an assessment of the equipment conditions, safety, and adherence to operating procedures and regulatory requirements. The following areas were among those inspected:

-- Turbine Building;

-- Vital Switchgear Rooms;

-- Cable Spreading Room;

-- Diesel Generator Building;

-- Reactor Building; and,

-- Battery Rooms.

The following items were observed or verified:

a. Fire Protection:

-- Randomly selected fire extinguishers and hose stations were accessible and inspected on schedule.

-- Fire doors were unobstructed and in their proper position.

-- Ignition sources and combustible materials were controlled in accordance with the licensee's approved procedures.

-- Appropriate fire watches or fire patrols were stationed when equipment was out of service.

-- Fire retardant wood was used for scaffolding.

b. Equipment Control:

-- Jumper and equipment mark-ups did not conflict with Technical Specification requirements.

-- Conditions requiring the use of jumpers received prompt licensee attention.

-- Administrative controls for the use of jumpers and equipment mark-ups were properly implemented.

-- Breakers for electrical equipment being worked were properly tagged out.

c. Vital Instrumentation:

-- Selected instruments appeared functional and demonstrated parameters within Technical Specification (Tech Spec) Limiting Conditions for Operation.

d. Housekeeping:

-- Plant housekeeping and cleanliness were in accordance with approved licensee programs.

No concerns were identified.

1.3 A problem with the operability status of the fire pumps continued during this report period. The types of problems included out of specification low RPMs with the diesel and water leaks from the pump relief valve and piping connections. These are recurring problems and have resulted in Plant Engineering initiating a Tech Functions Work Request to address them. Tech Functions was evaluating the problems at the end of the report period. The fire water pumps are required by Tech Specs; the licensee has followed appropriate Action Statements.

1.4 The inspectors observed receipt, handling, QC inspection, and storage of new fuel assemblies. No concerns were identified.

1.5 The 'C' and 'D' Emergency Service Water (ESW) pumps were determined to be in the ASME Code Section XI Action Range for flow during performance of the routine monthly surveillance/IST. They were declared inoperable. Troubleshooting concluded that the problem was not with pump performance but with the flow measuring instrumentation. As a result the "Controlatron," an ultrasonic flow measuring device, was repaired and the IST reperformed. The results indicated 'C' pump performance was within acceptable limits, but that the performance of 'D' pump was still in the Action Range. 'C' pump was declared operable and 'D' remained inoperable. Plant Engineering reviewed other monitored pump and system parameters and concluded there was nothing wrong with the performance of 'D' pump and subsequently re-established new baseline data and declared the pump operable. The inspectors expressed a concern with the practice of establishing new baseline data because it is done on a fairly frequent basis for the ESW pumps and appears inconsistent with the purpose of Section XI IST. The inspectors will review the ESW IST program in a subsequent inspection.

2. Observation of Physical Security

During daily tours, the inspectors verified access controls were in accordance with the security plan, security posts were properly manned, protected area gates were locked or guarded, and isolation zones were free of obstructions. The inspectors examined vital area access points to verify that they were properly locked or guarded and that access control was in accordance with the security plan.

The licensee invited the inspectors to attend a vendor presentation involving state of the art motion detection systems. The inspectors attended the presentation and found it informative. The licensee is continuing to pursue upgrading their present motion detection system to, in part, eliminate nuisance alarms.

No concerns were identified.

3. Radiation Protection

During entry to and exit from the RCA, the inspectors verified that proper warning signs were posted, personnel entering were wearing proper dosimetry, personnel and materials leaving were properly monitored for radioactive contamination, and monitoring instruments were functional and in calibration. Posted extended Radiation Work Permits (RWPs) and survey status boards were reviewed to verify that they were current and accurate. The inspector observed activities in the RCA to verify that personnel complied with the requirements of applicable RWPs and that workers were aware of the radiological conditions in the area.

On December 17, 1985 nine workers were slightly contaminated while working in the Augmented Offgas (AOG) building. The radionuclides involved were short-lived rubidium-88 and cesium-138. The airborne contaminants were apparently released into the AOG atmosphere when the end bell was removed from the 'B' recombiner blower during blower troubleshooting activities. Prior to removing the end bell, the 'B' recombiner subsystem (blower and palladium catalyst bed) had been isolated and purged with clean air. The licensee concluded the radioactive contaminants entered the 'B' recombiner subsystem through leaking valves. Corrective action was taken by the licensee to modify procedures and valve lineups when isolating a recombiner subsystem for maintenance to help preclude recurrence of this problem.

At the end of this report period, the AOG system was back in service after completion of helium leak testing and repairs of identified leaks. One source of gas leaks from the system into the building atmosphere was determined to be through cracks in stainless steel hydrogen analyzer instrument piping apparently resulting from intergrannular stress corrosion cracking. The affected piping was replaced. Not all leaks were identified as indicated by the presence of radionuclides in the AOG building during subsequent system operation. The indicated radiation levels, however, have decreased from previous levels as a result of the repairs.

4. Expanded Safety System Facility (ESSF)

Vibroflotation of the fill adjacent to the north side of the reactor building was in progress to prepare the area for the ESSF. As a result of this fill compaction activity, unexpected ground settlement occurred and the vibroflotation was stopped. The licensee and their consultants were involved in analyzing the impact of the unexpected settlement on the ESSF at the end of the report period. The licensee agreed to make a technical presentation to NRC Licensing by the end of March 1986 to discuss the impact of this settlement on previously discussed issues.

5. Status of Heating, Ventilating, and Air Conditioning (HVAC)

At the request of the inspectors, the licensee explained to them the history of HVAC problems at Oyster Creek and the efforts completed, in progress, and planned to restore and upgrade plant HVAC systems. The inspectors requested this briefing because of the HVAC problems occurring at the plant. Although most of these HVAC systems are not safety-related, their proper functioning is important in providing plant habitability, maintaining equipment operating environments within design, and controlling the spread of radioactive contamination.

From the briefing, it was learned that many of the originally installed HVAC systems had seriously degraded to the point that they were non-functional. Additionally, there were cases of inadequate design. In the early 1980's efforts were initiated to correct the problems and many systems have since been repaired and upgraded to a functional status in the manual mode with restoration of automatic capability planned for the future. Degraded items that have been repaired or are in the process of being repaired include ductwork, insulation, supports, dampers, automatic controls, rotating equipment, protective enclosures, piping, heating and cooling coils, flexible joints, filters, air operators, actuators, and control circuitry. Preventive maintenance programs and periodic inspections are now in place to maintain an operational status. The inspectors concluded from this briefing that the licensee is aware of the importance of HVAC and has spent a considerable amount of effort in upgrading and restoring degraded equipment. The number and type of HVAC problems that have occurred over the past year are indicative, however, that certain repairs were not sufficiently scoped and/or effected and that problems will continue to arise and require attention.

There appears to be sufficient management attention regarding HVAC to provide confidence that the major HVAC systems will be totally restored and maintained.

6. Hydraulic Snubber Operability

A review of Short Forms (work orders) involving work on two snubbers indicated a violation of Tech Spec requirements had occurred. Short Form 26521 involved repairs to non-Tech Spec snubber 15/1. The repair of snubber 15/1 involved replacement of a damaged paddle. When a spare paddle could not be located, it was determined the paddle on snubber NQ-2-S7 would be suitable for use on 15/1 and that a spare paddle could be used on NQ-2-S7. This resulted in the issuance of SF 26522 to remove the attached front paddle on NQ-2-S7 and to replace it with a spare. The Tech Specs state in paragraph 3.5.A.8.a, Shock Suppressors (Snubbers):

> During all modes of operation except cold shutdown and refuel, all safety related snubbers listed in Table 3.5.1 shall be operable except as noted in 3.5.A.8.b, c and d below.

Paragraph 3.5.A.8.b states:

> From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

Snubber NQ-2-S7 is included in Table 3.5.1 and 15/1 is not. Paragraphs 3.5.A.8.c and d do not permit the licensee to vary from the requirements of 3.5.A.8.a and b. The action of disconnecting and exchanging parts on a Tech Spec snubber to enable repair of a non-Tech Spec snubber rendered the Tech Spec snubber inoperable. This action violated the above Technical Specification in that snubber NQ-2-57 was previously operable and had not been determined inoperable prior to paddle replacement. This is a violation (219/86-02-01). The snubber was inoperable for less than 72 hours, thus, despite the unacceptable action to degrade a fully operable snubber, no plant shutdown was required.

A review of the particular circumstances that led to this violation indicated SF 26522 was not clearly written and led the Group Shift Supervisor to believe the Tech Spec snubber was inoperable to begin with. As a result, he approved the work activity. Additionally, Station Procedure 775.1.004, Rev. 9, Removal/Replacement of Bergen-Paterson Hydraulic Snubbers, did not address control of snubber removal. These issues should be addressed by the licensee in their response to this violation.

During NRC review of this item, the licensee pointed out that a proposed Tech Spec Amendment, now under NRC review, will slightly revise the wording so that voluntarily rendering a snubber inoperable will be permissible, so long as the Action Statement time limit is not exceeded. The inspector reviewed the basis for the provision and found that it is to allow preventive maintenance, testing and minor corrective maintenance to improve equipment reliability. Even under the proposed Tech Spec wording, scavenging parts from a fully operable snubber would be considered an abuse of the Tech Spec, and would not be considered acceptable.

7. Meetings with Management

The inspectors met with senior management of Maintenance, Construction, and Facilities at which time they were updated on key personnel changes both onsite and in Parsippany. Additional subjects discussed included:

-- Improving efforts to anticipate the need for Tech Spec changes during job planning;

-- Expanding the job monitor program to use contractor personnel;

-- SALP report and response; and

-- Planning for the 11R outage.

The inspectors also met the the Chairman of the General Office Review Board who is also responsible for the Independent Onsite Safety Review Group. The topics discussed included the safety review, Tech Spec change, and unreviewed safety question processes.

8. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification requirements were reviewed by the inspectors. This review included the following considerations: the report includes the information required to be reported to the NRC; planned corrective actions are adequate for resolution of identified problems; and the reported information is valid. The December, 1985 Monthly Operating Report was reviewed by the inspectors.

In addition Special Report 85-03, involving deficiencies with 14 fire dampers, was reviewed during this report period. The inspectors were interested in this because the deficiencies were identified as a result of licensee followup of NRC Information Notice 83-69. The inspectors expressed a concern regarding the two year delay in followup. The licensee stated efforts were in progress to catch up with the backlog. The inspectors have observed licensee activities that indicate they are aware of and addressing the backlog problem.

No concerns were identified.

9. Reactor Water Level Instrumentation

Low reactor water level sensors were replaced during the cycle 10M outage (October 18 to November 16, 1985) to comply with the Environment Qualification Rule (10 CFR 50.49). The previous sensors, equipped with indicating gauges, were replaced with environmentally qualified switches without indicating gauges. This became significant when the Tech Spec requirement to perform a channel check on the instruments to verify hydraulic communication of the sensor with the reactor was considered. The Office of Nuclear Reactor Regulation (NRR) required that an alternate means be devised to verify that the sensor was in hydraulic communication with the reactor vessel after performing routine surveillance actions. This would provide assurance that the instrument was correctly returned to service after performing the required surveillance actions, similar to the function the indicating switches previously served. The licensee, to answer NRR's requirement, proposed a special valving sequence be performed to provide assurance that the instrument was in hydraulic communication with the reactor vessel. The manufacturer, Static-O-Ring, Inc., concurred that this special valving sequence would pose no adverse affect on the switches nor instrument setpoint recovery.

During a routine surveillance test on January 17, Reactor Water Low Level Scram Sensors RE 05/19A1, RE05A1, and RE05/19B1 setpoints were found to have drifted out of specifications. The instrument setpoints were corrected and the sensors returned to service. On January 20, the surveillance test was repeated to determine if the instrument drift experienced on the 17th may have been caused by the special valving sequence employed when returning the instruments to service following a surveillance. During the repeat surveillance test on RE05A1, a half scram signal, initiated as part of the surveillance, could not be reset. The licensee commenced a reactor shutdown and subsequently was able to clear the half scram by repeating the surveillance on RE05A1, but not before the Tech Spec requirement allowing an instrument to be inoperable for one hour per month for testing, was exceeded by eight minutes. The reactor shutdown was halted after the surveillance was completed until the licensee could make a determination regarding the operability of the instrument. Based on the erratic performance of RE05A1, the licensee decided to declare the instrument inoperable and recommenced a reactor shutdown. Plans were immediately formulated to replace the instrument during the reactor shutdown. The new instrument was installed, calibrated, and returned to service and as a result the shutdown was halted at 500 MWE.

The licensee is conducting an investigation to determine the cause of the instrument failure, including having the manufacturer conduct a failure analysis of the instrument. Preliminary findings indicate that the failure was a result of high differential pressure across the instrument, as a result of the special valving sequence performed during surveillance testing. The licensee has developed another method to ensure the instruments are in hydraulic communication with the reactor vessel which involves the installation of test isolation valves on the high and low pressure vents on each instrument. These valves were installed on RE05A1 when the switch was replaced. This new method will significantly reduce the differential pressure that the sensor diaphragm experiences during surveillance testing. The licensee has developed a modification package to install the test isolation valves on the remaining low level instruments as they are surveilled.

10. IRM Number 10 Safety Evaluation

This concern was previously addressed in item 219/85-23-07 as a result of the MSIV closure scram on June 12, 1985. During the scram, the operators were unable to reset the trip until the reactor was depressurized to the 600 psig bypass setpoint for MSIV closure and the low condenser vacuum scram. The 600 psig bypass setpoint allows the operator to establish a main condenser vacuum and to supply steam to the secondary plant. After the 600 psig setpoint is exceeded, the reactor protection circuitry places the MSIV closure and low condenser vacuum scram in service. Therefore, decreasing plant pressure to 600 psig disables the MSIV scram and allows the trip to be reset. The inability to reset a scram until the plant is depressurized to 600 psig, when a valid scram condition no longer exists, is a concern. During the June 12, 1985 scram, this inability coupled with the failure of the scram discharge volume (SDV) drain valves to seat properly, resulted in a steam release to the reactor building.

An analysis conducted by General Electric for BWR 5 and 6 series reactors permitted raising the bypass setpoint from 600 psig to 800 psia. This was submitted by the licensee and approved as a Tech Spec change. Due to other considerations that follow, the plant elected to continue to operate with the bypass setpoint at 600 psig. If the licensee elects to change this setpoint to 800 psia and operate the plant in accordance with the change as authorized by Tech Specs, then an analysis or documentation should be reviewed to ensure the General Electric BWR 5/6 analysis is suitable for the Oyster Creek facility. In addition, the basis for Tech Spec Section 3.1, Protective Instrumentation (page 3.1-3a) uses 600 psig while Table 3.1.1 Note B (page 3.1-12) uses 800 psia; this should be clarified. The licensee should adequately address the MSIV closure scram reset point (600 psig) in their analysis, clarify the Tech Spec with regard to the bypass setpoint, and provide documentation to support the Technical Specification analysis that changed the setpoint from 600 psig to 800 psia. These items are unresolved. (219/86-02-02)

The 600 psig bypass setpoint was not changed to the 800 psia setpoint as authorized by Tech Specs due to a concern the licensee developed as a result of the Intermediate Range Monitor (IRM) System Range 10 modification. The IRM Range 10 modification was installed during the cycle 10 refueling outage to facilitate the clearing of LPRM downscales during reactor startups. The safety evaluation conducted to support their modification was completed in May 1982, but additional concerns were raised by the licensee in June 1984, after NRC:NRR acceptance of the licensee's safety analysis for the modification, concerning the acceptability of this modification. The licensee's concern involved a postulated reactivity addition accident (excessive feedwater addition, idle recirculation loop startup, etc.) occurring during a reactor startup while in the intermediate range.

The amount of reactivity addition was theorized to be sufficient to exceed the 25% safety limit associated with the GEXL correlation and possibly causing core damage. The scenario would have the operator, when less than 600 psig, range the IRMs to range 10 (in violation of plant procedures) when power escalated with the reactivity addition. Operator upranging to IRM range 10 would place a scram setpoint of 38.4% in effect and, coupled with plant pressure below 600 psig (which bypasses the MSIV closure scram initiated when switching to range 10 from below 850 psig), could potentially lead to exceeding a safety limit. The IRM 10 safety evaluation discussed the successful transfer to IRM Range 10 when reactor pressure was greater than 850 psig and the protection afforded by an MSIV closure scram if the transfer was attempted below 850 psig. This is not entirely correct as the safety evaluation did not consider transferring to IRM range 10 from below 600 psig which may be conducted without an MSIV closure scram. Corporate engineering wrote an internal memorandum (August 1984) addressing this concern stating that no fuel damage would occur, but did not properly address exceeding the safety limit. The memorandum also offered to further analyze the event and explore solutions to eliminate inadvertent entry into IRM range 10. The licensee had taken no action in this area until a recent inadvertent entry into IRM range 10 caused an MSIV closure scram to occur during a scram recovery. The licensee is currently contemplating hardware and circuitry changes as a possible solution if their analysis supports a modification. Pending the licensee's resolution, the potential to exceed the 25% safety limit as a result of the IRM Range 10 modification is an unresolved item. (219/86-02-03)

11. Reactor Building 23' Elevation Sprinkler

The automatic sprinkler system on the Reactor Building 23' elevation was actuated on January 14, 1986 by truck exhaust fumes. The truck was located in the reactor building access airlock, unloading new fuel canisters. The exhaust fumes caused an ionization detector in the fire protection system to initiate deluge system No. 8 (reactor building system south - cable trays at elevation 23'). The licensee had previously installed a ventilation exhaust trunk to safely conduct the exhaust fumes to the outside environment during return of spent fuel from West Valley. This arrangement was removed after the West Valley spent fuel shipments were completed and was not reinstalled for the new fuel shipments. The licensee has refabricated an exhaust trunk to remove the truck fumes and plans to reflect this in a procedure change.

The deluge system actuation wet safety related equipment and as a result Core Spray System II was declared inoperable by the licensee. Core Spray System II is equipped with splash shields to protect the vital equipment from this type of event. The pump motors were meggared and examined to determine if there was any degradation. The test results indicated that

Core Spray System II was unaffected by the event and it was, therefore, returned to service. The licensee performed additional walkdowns of the 23' elevation to determine if any equipment had suffered damage. No problems were identified. A fire watch was established until the deluge system was returned to operation.

No inspector concerns were identified.

12. Fuel Pool Liner Repair

The licensee discovered leakage from the spent fuel pool liner coming from the tell-tale drain lines in the Shutdown Cooling room in December. The initial leak rate was approximately three gallons per hour. The licensee employed a helium leak test to locate the leak in the spent fuel pool next to a swing bolt used to fasten previous spent fuel racks. The licensee speculated that the cracks in the liner resulted from relieving the stresses placed on the liner by the torqued swing bolt and the weight of the spent fuel rack. It was at this time when the old fuel rack was being removed from the pool that the licensee noticed the leak in the liner as indicated by the flow of water from the tell-tale drain.

Repair procedures were developed and a diving contractor was selected to perform the underwater weld repair. The repair involved fillet welding a 10" diameter piece of pipe, approximately 8" long with a cover plate on one end, to the liner floor. This pipe section covered the swing bolt and the cracks, thus isolating the leaks. The licensee took special precautions to minimize the diver's exposure, including relocating items in the pool. The inspector met with onsite Radiological Controls personnel to discuss licensee planning and preparation for the repair of the Spent Fuel Pool liner.

The following matters were discussed by a Region I based Radiation Specialist on January 30, 1980:

-- planning and preparation,

-- establishment and approval of procedures (as necessary) for:

    -- diving operations
    -- emergency response (e.g. loss of breathing air, loss of pool water, damage of diving equipment and suits)
    -- exposure control including source checking radiation survey instrumentation

-- pool decontamination,

-- radioactive source control (e.g. incore instrumentation),

-- control of access to fuel,

-- licensing requirements (if necessary),

-- training of personnel in changes to procedures,

-- water clarity,

-- control of diver approach to spent fuel,

-- dose mapping of pool including gamma and neutron radiation,

-- personnel dosimetry and its calibration (whole body, skin, and extremity),

-- use of multiple survey instrumentation, its calibration and periodic verification of operability,

-- use of survey meters and alarming dosimeters during underwater work,

-- contamination control including control of possible point sources (e.g. small chips),

-- control and verification of movement of spent fuel,

-- bioassays of diving personnel,

-- training and qualification of personnel on applicable procedures,

-- applicable NRC guidance in this area (e.g. IE Information Notice No. 84-61, "Overexposure of Diver in Pressurized Water Reactor (PWR) Refueling Cavity"),

-- breathing air quality for divers,

-- previous diving operations at reactor facilities in NRC Region I.

Documents Reviewed

-- Procedure A15A-51752, Rev. 0, "Underwater Repair of Spent Fuel Pool Liners," January 23, 1986

-- Procedure MTH-80-0004, Rev. 6, "Procedure for Underwater Diving Work Associates," dated December 7, 1983 (Nuclear Utility Construction)

-- ALARA Review 86-047, "Perform Weld Repair of Spent Fuel Pool Liner," Rev. 2

-- Various underwater radiation survey results

-- personnel training records

-- applicable Radiation Work Permit

Findings

Within the scope of this review, the following matters needing licensee attention were identified. The licensee satisfactorily resolved these issues prior to inspection completion of reviews in this area.

-- Evaluate adequate calibration of personnel monitoring devices (e.g. TLDs, pocket dosimeters) used to quantify personnel exposure in the spent fuel pool. Ensure the calibration is appropriate for the radiological environment (e.g., gamma, and beta radiation).

  The licensee evaluated the energy monitoring capabilities of his personnel monitoring device and determined it to be acceptable. Algorithms could properly quantify exposure.

-- Evaluate the capability of radiation survey instrumentation (meters, pocket dosimeters) to properly assess exposure conditions in the spent fuel pool. Ensure instrument calibration is acceptable for the radiological environment being monitored.

  The licensee evaluated the capabilities of his instruments and determined them to be acceptable.

-- Provide/ensure clear guidance relative to personnel that are responsible for ensuring workers are trained (qualified in diving procedures).

  The licensee revised procedures to address this matter.

-- Establish clear guidance on actions to take following potential identification of damage to diving equipment and suits.

  The licensee revised procedures to address this matter.

-- Provide controls and establish a minimum program (as necessary) to bioassay and determine a potential intake of tritium and alpha emitters.

  The licensee performed an appropriate evaluation and reviewed controls to ensure these matters were addressed.

The licensee agreed to address and resolve the following matters prior to diving operations (50-219/86-02-04).

-- Evaluate potential neutron doses to divers,

-- Clearly define minimum requirements for source/response checking of survey instruments and alarming dosimeters prior to each dive,

-- Provide specific guidance relative to minimum acceptable water clarity for diving operations to start/continue,

-- Establish diving suit contamination/radiation limits, and

-- Provide controls (as necessary) to ensure the use of proper instrumenttation for performing underwater surveys by a diver.

The following positive attributes were noted:

-- The licensee performed and documented an ALARA review for the diving operation.

-- The licensee decontaminated the work area by use of underwater vacuuming.

-- Water clarity was very good.

-- The licensee removed radioactive sources from the work area, including spent fuel.

-- The licensee assigned and dedicated specific individuals to the task.

As a result of the licensee's effort, the diver received relatively low exposures; a whole body dose of 29 mrem and an extremity dose of 94 mrem was received during the 3 hour and 18 minute dive.

13. Surveillance Testing

The inspector reviewed the following surveillance tests to determine if the tests were included on the master surveillance schedule, were technically adequate, and were performed at the required frequency.

619.3.013 -- Reactor Low Level Test and Calibration, Revision 17, 12/16/85

This surveillance was performed on January 20, 1986 and is discussed in paragraph 9. Short Form 33076 was written to replace RE05A1.

665.3.021 -- Containment Electrical Penetration Nitrogen Blanket Surveillance, Revision 2, 11/08/85

636.4.003 -- Diesel Generator Load Test, Revision 23, 12/02/85

645.2.002 -- Fire Pump Diesel Battery Verification, Revision 12, 9/26/85

620.3.003 -- APRM Surveillance Test and Calibration, Revision 13, 9/20/85

Short Form 32063 was written to have the power test potentiometer cleaned or replaced.

14. Licensee Action on Previous Inspection Findings

(Closed) Inspector Followup Item (219/85-23-07): Adequacy of IRM Range 10 modification to insure 25% safety limit is not exceeded during a startup reactivity accident.

This item is updated in this report (paragraph 10) and was changed to two unresolved items. Licensee action will be tracked by the unresolved items; therefore, this item is closed.

15. Exit Interview

A summary of the results of the inspection activities performed during this report period were made at a meeting with senior licensee management at the end of the inspection. The licensee stated that, of the subjects discussed at the exit interview, no proprietary information was included.