U. S. REGULATORY COMMISSION
REGION I

Report No. 50-219/87-42

Docket No. 50-219

License No. DPR-16 Priority -- Category C

Licensee: GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, New Jersey 07054

Facility Name: Oyster Creek Nuclear Generating Station

Inspection Conducted: December 20, 1987 - February 6, 1988

Participating Inspectors: J. Wechselberger, Senior Resident Inspector
E. Collins, Resident Inspector

Approved By: C Cowgill III 3/14/88
C. Cowgill, Chief, Reactor Projects Section 1A Date

Inspection Summary:

Areas Inspected: Routine inspections were conducted by the resident inspectors and one region-based inspector (151 hours) of activities in progress including operations, radiation control, physical security, surveillance and maintenance. The inspectors also periodically toured the control room and other portions of the plant and reviewed periodic and special reports. In addition the inspectors participated in selection of fasteners for Bulletin 87-02, reviewed licensee activities surrounding the freezing of instrument lines and examined feedwater flow nozzle and core thermal power calculations. The inspection period included review of operational considerations including high pressure scram surveillances (RE03), "C" main battery, "C" EMRV acoustic monitor and safety valve thermocouples, as well as the review of two other industry problems for applicability to Oyster Creek; specifically a feedwater regulation valve failure and diesel engine exhaust manifold combustion. Additionally, the inspectors reviewed inspection activities associated with snubber visual inspection surveillance.

Results: No Violations were identified. The high pressure scram instruments (RE03) continue to exhibit setpoint repeatability problems. A special inspection (50-219/88-02) was conducted to review the instrument line freezing event. The licensee plans to conduct additional noise signature recording of "C" EMRV acoustic monitor in an attempt to determine if the signature is representative of a steam leak or flow noise. The review of the industry diesel problem determined that this particular problem was very unlikely to occur at the facility. The licensee determined that the feedwater valve cyclic fatigue problem had never occurred at the facility, but elected to include this as an inspection attribute during the next outage.

DETAILS

1.0 Freezing Instrument Lines in Reactor Building

On 1/6/88, #2 seal pressure indicator for the "B" and "E" recirculation pumps and the shutdown cooling pumps discharge pressure instrument lines showed indications of freezing. The seal pressure increased from approximately 520 psig (normal pressure indication) to about 800 psig. The operators initially suspected a seal failure, but later determined that the instrument lines were freezing as a result of cold reactor building temperatures. The operators measured the temperatures on the 23' elevation in the area of the freezing instrument lines at approximately 33°F and as a result the operators took action to reduce the cold air induction into the reactor building by securing normal reactor building ventilation and initiating the standby gas treatment system (SBGTS). As a result within approximately 30 minutes the recirculation seal pressures had recovered.

The inspectors discussed the event and toured the reactor building with the licensee to conduct temperature surveys and to assess the potential impact of the freezing temperatures in the reactor building on safety systems. Building temperature surveys conducted with SBGTS running indicated acceptable temperature surveys throughout the building. Temperature surveys conducted after normal building restoration, including repair of the heating coil, again, yielded acceptable results.

Cold building temperatures were a result of the reactor building heating coil being out of service in need of repair. The licensee was able to repair the coil and restore normal building ventilation by late afternoon on 1/6/88. The inspector questioned the licensee about the repair and why it wasn't accomplished before 1/6/88 when the licensee had experienced cold building temperatures in November. The licensee explained that the heating coil work was of a lesser priority than more important work. The licensee continued to experience a problem in maintaining the ventilation heating coils as a result of flow control valve malfunctions. The inspector identified and discussed additional concerns with the licensee which are detailed in Inspection Report 88-02.

2.0 Core Thermal Power Calculation

The licensee notified the resident inspectors that there was a concern that for some period of time the reactor may have been operated at power levels above 1930 megawatts thermal (MWt), the licensed limit. This concern was generated by a review of plant operating parameters which prompted a calibration check of the feedwater temperature instrumentation. This calibration check indicated that all components were within the specified tolerances; however, the millivolt/current converter was slightly readjusted to increase the accuracy at the operating point. When the temperature loop was returned to service, it was noted that core thermal power, as indicated by the process computer, has increased from 1922 MWt to 1938 MWt. This indicated increase in reactor power level was attributed to the adjustment of the millivolt/current converter as other plant parameters had not changed. Reactor power level was lowered to below 1930 MWt indicated.

The resident inspectors met with licensee personnel on 1/8/88 to discuss feedwater temperature calibration. It was concluded that while the temperature calibration methods included each individual component, it was also necessary to provide a loop calibration check from the source to the computer indication, with specified tolerances. The licensee stated that it was their intention to generate a procedure to accomplish this.

As part of the core thermal power calculation review, detailed reviews of the feedwater flow calibration procedure 1001.30 (Reactivity Measurements - Periods) and Technical Data Report (TDR) #842, were performed. The inspector concluded that while the methods and theoretical basis for the calibration were sound, uncertainties in actual feedwater flow could result from changes in the physical characteristics of the flow elements due to erosion or fouling during the period since installation. Also, TDR-842 indicated that the feedwater flow nozzles were not calibrated prior to installation, making it impossible to verify the nozzle discharge coefficients used in procedure 1001.30. Furthermore, the differential pressure transmitter instrument line taps were modified, but never evaluated. A similar modification at Brown's Ferry resulted in almost a 1% change in the nozzle discharge coefficients. All of these factors taken together serve to introduce uncertainty in the feedwater flow input to the core thermal power calculation. In addition, feedwater nozzle coefficient may change with time due to either erosion or corrosion and the nozzles are not routinely calibrated to a known standard which would identify any changes in the nozzle coefficient. The inspectors questioned the licensee if the above identified uncertainties associated with measuring feedwater flow could be bounded within the allowable uncertainty for core thermal power calculations.

Additionally, a review of several reactor plant parameters, panel steam flow, panel feedwater flow, computer steam flow, computer feedwater flow, and condensate flow showed poor agreement. Specifically, there is a large mismatch between computer feedwater flow and all other similar indications of the same parameter. This concern was expressed to the licensee who initiated maintenance actions to determine if any steam flow discrepancies existed.

The inspectors will continue to evaluate the feedwater flow conditions and its effect on core thermal power.

3.0 Loss of Process Computer

On 1/30/88 at approximately 5:15 PM, the SIGMA process computer failed and efforts were initiated to restore the computer to operation. The operation of the process computer is significant because it uses plant parameters to calculate the core thermal power that operators use to ensure compliance with the licensed limit of 1930 Megawatts thermal (MWt).

At approximately 9:45 PM the same day, the control room operators performed a manual heat balance using the guidelines of Station Procedure 1001.6, Core Heat Balance - Power Range, Revision 9. The calculated core thermal power using this procedure was 1964.7 MWt. Previously the inspector had questioned the licensee concerning the accuracy of procedure 1001.6 in the event that a manual heat balance would be required. Apparently, the feedwater flow indication used for this computation had drifted from its initial calibration and could not be corrected until more flow data points were obtained. This evolution would require significant reductions in plant operating power levels.

It was decided, after contacting the plant operations manager and the core-manager, to maintain steady state power operation using alternate plant indicating parameters. The parameter specified to maintain was the electrical output of the generator (660-665). These levels are commensurate with the levels observed prior to the loss of the process computer.

Upon change of shift, at approximately midnight, the relieving shift performed another manual heat balance (1960 MWt) and expressed concern that the process computer was unavailable and that the manual heat balance contained errors. The plant operations manager was again contacted and it was agreed to reduce plant power to provide margin to the license thermal limit. Plant power was reduced by approximately 20 MWt and a new manual heat balance yielded 1938.75 MWt. As a check of actual reactor power, the Group Shift Supervisor (GSS) used the feedwater calibration procedure to calculate feedwater flow and used this as the feedwater flow input to the manual heat balance. Subsequently this method was incorporated as a change to the manual heat balance procedure, 1001.6. The GSS was satisfied that the reactor was below its license thermal limit.

The process computer was returned to service at approximately 4:45 AM on 1/31/88.

The inspector is concerned that the feedwater flow millivolt (MV) indicator had a known deficiency and yet this was not formally documented in the control room. The manual heat balance instruction, 1001.6, clearly specifies the use of this indicator, yet the control room operators had no confidence in it. Without the process computer, there was no direct indication of core thermal power available to the operators.

Prior to the loss of the computer the inspectors asked core engineering what operator actions would be required if the deficient 1001.6 procedure was used. The core engineers stated that the requirements of the procedure would be followed even though it was considered deficient. Initially, operators choose to maintain core thermal power constant as no plant conditions had changed. Subsequently, though a 20 MWt power reduction was initiated after a shift change had occurred. The oncoming shift supervisor expressed concern at exceeding the manual heat balance thermal limit and reduced power. The inspectors determined that there was no immediate safety concern as no actual thermal power limit was exceeded, but were concerned with regard to the lack of action to identify to the operators the deficiencies and lack of direction in specifying operator actions in this situation. The licensee acknowledged the inspectors concerns and as a result planned to review the event.

4.0 Review of NRC Bulletin

Bulletin No. 87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications

NRC procurement inspections in the past have included the testing of a sample of fasteners. These sample results have indicated the need for a larger sampling basis and resulted in the current effort to request licensee's select according to usage and test fasteners to determine if specification requirements for mechanical and chemical properties are being met.

The inspector observed the licensee's sampling of safety and non-safety related bolts, studs or screws and nuts. A large portion of the licensee fasteners are ASTM A193 grades B7 and B8. The licensee did not have usage data immediately available; therefore the selected fasteners are all ASTM A193 with one exception. The exception is indicated in the attachment to this report. All nuts selected were ASTM 194. In all cases a minimum of 2 of each item was selected to support the laboratory testing for mechanical and chemical properties. The inspector verified that the samples were properly tagged to ensure traceability of the sample to the sample data sheet. Attachment 1 to this report is a description of the sampling of fasteners selected.

5.0 Plant Operational Review

5.1 The inspector reviewed details associated with key operational events that occurred during the report period. As summary of these inspection activities follows.

-- On 1/13/88 the licensee experienced a "C" condenser vacuum transient which resulted in a vacuum drop from approximately 29 inches HG to approximately 26 inches HG in the "C" condenser. The operators recognized the problem and took actions to avert a possible plant shutdown. Initially large oscillations were experienced in the off gas line flow and operators took action in response to this indication. They then recognized the potential for the steam traps for the "C" inter or after steam jet air ejector (SJAE) condenser to be sticking shut. When these steam traps, Y-4-001 and Y-4-007, stick shut, water is not removed from the SJAE condensers and results in a decrease in vacuum. Subsequently the steam traps have periodically stuck resulting in a vacuum decrease and required operator response to free the traps. The inspector questioned the licensee if a repair could be effected to prevent the steam traps from sticking shut. The licensee presently plans a power reduction to perform an unrelated surveillance and at that time performing maintenance on the steam traps.

-- The resident inspector examined the Oyster Creek emergency diesels for susceptibility to combustion in the exhaust manifold. This review was performed in response to a reported problem with opposed piston diesel configurations. Apparently, oil remains on top of the upper piston after diesel engine operation and, upon shutdown, is able to seep into the cylinder area and into the exhaust manifold. On subsequent engine operation, the oil in the exhaust manifold, if sufficient oxygen is present, can ignite. A review of the diesel configuration at Oyster Creek showed that the diesel configuration is of the 'V' type vice the opposed piston type. It is considered unlikely that this phenomena, as described, could occur in this diesel configuration. Also, discussions with the licensee indicated that previous inspections of the exhaust manifold and the blower diffuser screen showed no evidence of oil accumulation. The inspector has no concerns in this area.

-- During the report period the licensee initiated several maintenance actions to correct discrepancies with hydraulic control accumulators (HCU). These items included level switch problems on HCU 10-11, immediate maintenance to correct a leaking V-111 on HCU 38-19, replacement of valve 118 on HCU 26-47 and work on the "48" notch indication of the position indication probe on HCU 14-47 which was determined to a be drywell problem. The inspector will continue to follow the number of HCU problems.

-- During this period the thermocouple for main steam relief valve NR-28H was declared inoperable as result of erratic indication. The inspector verified that in accordance with technical specifications the gain on the adjacent acoustic monitor, NR28J, was increased. In addition to NR-28H, thermocouples for NR-28A and NR-28C had already been declared inoperable and the gain on the adjacent acoustic monitors increased.

5.2 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.

5.3 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 are among those inspected:

-- Turbine Building

-- Vital Switchgear Rooms

-- Cable Spreading Room

-- Diesel Generator Building

-- Reactor Building

The following additional items were observed or verified:

a. Fire Protection:

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

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

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

b. Equipment Control:

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

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

c. Vital Instrumentation:

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

d. Housekeeping

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

No inspector concerns were identified.

6.0 RE03's High Reactor Pressure Switch

During this report period, the inspector reviewed the failure of RE03B to trip as required during surveillance procedure 619.3.017, "Reactor High Pressure Scram Test and Calibration," on 12/17/87. The licensee determined the microswitch to exhibit poor repeatability and placed the second microswitch in service. During the next surveillance on 1/17/88 the second microswitch in RE03B failed the surveillance test and again exhibited poor repeatability. This repeatability problem was slightly different from the first RE03B failure in that it exhibited double trip point indications. The licensee placed a 1/2 scram in the reactor protection system and replaced the failed microswitches on RE03B. The instrument successfully passed the surveillance after switch replacement. The failed microswitches were sent to the licensee laboratory for analysis.

Earlier the RE03A microswitches failed during a surveillance and were replaced. The licensee conducted an analysis on the failed microswitch which indicated that carbon residue from apparent electrical arcing was found on the contact surfaces. The licensee believes that for the long period of operation the switch has been in service that the electrical arcing would occur as a result of the contact almost being made up as result of the close tolerances required by the switch setpoint.

The licensee is currently evaluating the switch performance and searching for a suitable instrument replacement. The inspector will continue to follow switch performance and the licensee's effort to find a suitable replacement.

7.0 Review of Nine Mile Point One Feedwater Transient Problem

On 12/19/87, Nine Mile Point One experienced a feedwater transient that resulted in a significant water hammer event. One of the root causes of the event was cyclic fatigue failure of the feedwater flow control valve stem and plug connection due to excessive vibration of the feedwater lines. The Nine Mile point flow control valve stem and disc had been fillet welded to correct early vibration problems.

The inspector reviewed the Nine Mile event with the licensee to determine applicability to the Oyster Creek facility. It was determined that both Nine Mile Point and Oyster Creek use a Fisher-Vulcan Valve. The licensee determined from a review of operating history that no event similar to the Nine Mile water hammer event had occurred at Oyster Creek. In addition, the licensee inspects the feedwater valve internals every outage and plans to accomplish the same activity when they shut down for the 12R outage. During the inspection, the licensee stated that they will inspect for signs of fatigue failure. The inspector had no further questions on this matter.

8.0 Snubber Visual Inspection

On 1/7/88 the inspector discussed results of a visual snubber inspection conducted under surveillance procedure, 675.1.001, Inspection of Bergen-Paterson hydraulic Snubbers with a licensee worker. The inspector learned that the baseplate associated with a core spray system snubber NF-2-S7 potentially had oversize bolt holes. Initially the material nonconformance report (MNCR) written on this problem was inspected by Quality Control and dispositioned as an optical illusion. The inspector discussed this with maintenance and quality control personnel. Apparently the quality control (Q.C.) people inspected the snubber baseplate (75' elevation) from the floor elevation level (51') with a video camera. The video camera inspection was not adequate to reveal the oversize baseplate holes; and therefore, Q.C. incorrectly dispositioned the MNCR. The inspector concluded that better communication could have existed between Q.C. and maintenance in resolving this nonconformance. In addition, the inspector questioned licensee personnel with regard to his perception of a reluctance of maintenance personnel to question Q.C.'s disposition of the MNCR. The maintenance people stated that there was no reluctance but that the supervisor was not able to immediately address the issue with Q.C. This was later confirmed by maintenance management.

The inspector noted that earlier inspections of the base plate had not identified the oversized holes.

As a result of the inspectors concerns, the licensee accompanied the inspector in witnessing a core spray surveillance test to determine if the baseplate was moving. The inspector observed no movement in the baseplate or excessive vibration in the core spray line. However, the licensee has recently changed the manner in which the core spray surveillance is conducted. In an effort to reduce potential water hammer problems with the core spray test line, the test valve is slowly jogged open which overall results in less system vibration. In addition during bulletin 79-14 inspection conducted in 1985 the bolts were found to be loose and had to be retorqued. The inspector concluded that adequate attention was not paid to inspection of the baseplate during previous inspections. The inspector will continue to review this area in future inspection activities.

In addition there was a question regarding the procedural requirement (675.1.001, paragraph 6.6.3.1) of a minimum thread engagement of 1/2 of the available thread length of the piston rod contacting the paddle threads between the piston rod and the monitoring paddle. The licensee changed the procedure to require full thread engagement instead of 1/2 thread engagement.

9.0 "C" Electromatic Relief Valve (EMRV)

The "C" EMRV acoustic monitor alarmed periodically on 1/25/88, but the downstream thermocouple temperatures gave no indication of an open EMRV. The operators reduced power to 1900 MWt in an attempt to clear the alarm condition. The power reduction was successful in clearing the alarm and also resulted in a slight decrease in the magnitude of the vibration. The alternate acoustic monitor was selected to determine if a malfunction existed particular to the primary acoustic monitor. The alternate acoustic monitor also exhibited the same behavior. In addition, the adjacent acoustic monitors exhibited the same acoustic behavior but displayed a lower magnitude. The licensee recorded the noise signature of the "C" EMRV for analysis. The inspector reviewed the analysis with the licensee. The licensee indicated that the noise signature frequency might be that associated with steam flow through a seal area or gasket or simply flow noise and not that of a pilot valve disc chatter. The inspector questioned if the EMRV had to be actuated if the potential existed that the EMRV would not be able to reseat as pressure may not be able to build up in the main disc chamber area to effect reseating. The licensee determined that any potential leakage past the pilot valve seat would be of small enough magnitude to allow the main disc to reseat. The licensee discussed this with the valve vendor and currently plans to have an outside contractor perform and analyze the noise signature of the "C" EMRV. The inspector will continue to follow the licensee efforts in this area.

10.0 (Open) Unresolved Item (219/87-33-01): "C" Main Station Battery Disconnected

The inspector conducted further review of this event concerning the alarm capability for the "C" main station battery. Annunciator "Bus C Input BRKRS (9xF-5-d)" open, alarms when either of the following occur, both "C" battery charger's output DC breakers are open or the "C" battery output breaker is open. The inspector reviewed Station Procedure 634.2.001, "Main Station Battery Discharge and Low Voltage Relay Annunciator Test," and discussed battery breaker lineup for the performance of the equalizing charge and test discharge of the 125 VDC battery with the licensee. The "C" battery breaker, in accordance with procedure, was open for the entire period, approximately 60 days, and thus the alarm window should have been lighted. The licensee conducted maintenance to determine if this annunciator was functioning properly during the event as the operators were not aware of the annunciator ever being in an alarm condition during the 60-day-period the "C" battery breaker was open. The licensee lifted a lead to the annunciator and received an alarm light in the control room. Presently the licensee suspects the contact switch in the "C" battery breaker malfunctioned and will test this at the first available opportunity. The inspector will review this effort.

11.0 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.

12.0 Observation of Physical Security

During daily tours, the inspectors verified that access controls were in accordance with the Security Plan, security posts were properly manned, protected area gates were locked or guarded and that 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.

13.0 Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification requirements were examined 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 following reports were reviewed:

-- Monthly Operating Report for December 1987.

-- Special Report 87-07 dated 11/19/87 concerning a non-functional fire barrier. A fire watch was established in accordance with the technical specification requirements until repairs were made.

-- Safeguards events log quarterly submittal dated January 20, 1988.

14.0 Instrumentation and Control Technician Concerns

The inspector reviewed the licensee's preliminary findings and recommendations regarding their review of some concerns raised by an instrumentation and control (I&C) technician. The report made recommendations for implementation by licensee management and concluded there were no nuclear safety concerns identified. The licensee presently plans to interview the remaining I&C technicians and to incorporate any additional findings in a final report to be issued on their review of this area. The inspector will review this report when it is completed by the licensee.

15.0 Exit Interview

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

ATTACHMENT 1

FASTENER SURVEY

ID#	SIZE	MAT'L SPEC.	DESCRIPTION
001	1/4-20UNC X 1.50	ASTM A193 B8M	SCREW HEX CAP
002	5/16-18UNCX 2.00	ASTM A193 D8M	SCREW HEX CAP
003	3/8-16UNC X 2.25	ASTM A193 B7	SCREW HEX CAP
004	7/16-14UNCX 1.00	ASTM A193 B7	SCREW HEX CAP
005	1/2-13UNC X 2.00	ASTM A193 B7	SCREW,HVY HEX
006	9/16-12UNCX 1.00	ASTM A193 D8M	SCREW,HVY HEX
007	5/8-11UNC X 6.00	ASTM A193 D8M	SCREW HVY HEX
008	3/4-10UNC X 2.00	ASTM A193 B7	SCREW HVY HEX
009	7/8-9UNC X 3.00	ASTM A193 B7	SCREW HVY HEX
010	1-8 UNC X 2.50	ASTM 193 D8M	SCREW HVY HEX

SAFETY RELATED NUTS

011	1/4-20 UNC	ASTM A194 8M	NUT HVY HEX
012	5/16-18 UNC	ASTM A194 8M	NUT HVY HEX
013	3/8-16 UNC	ASTM A194 2H	NUT HVY HEX
014	7/16-14 UNC	ASTM A194 2H	NUT HVY HEX
015	1/2-13 UNC	ASTM A194 2H	NUT HVY HEX
016	9/16-12 UNC	ASTM A194 8M	NUT HVY HEX
017	5/8-11 UNC	ASTM A194 8M	NUT HVY HEX
018	3/4-10 UNC	ASTM A194 2H	NUT HVY HEX
019	7/8-9 UNC	ASTM A194 2H	NUT HVY HEX
020	1-8 UNC	ASTM A194 8M	NUT HVY HEX

NON-SAFETY RELATED BOLTS, STUDS, OR SCREWS

021	1/2-13 X 2.50	ASTM A193 B8	SCREW, HEX CAP
022	5/16-18 X 1.50	ASTM A193 B7	SCREW, HEX CAP
023	1/2-13 X 1.25	ASTM A193 B8	SCREW, HEX CAP
024	5/8-11 X 4.00	ASTM A193 B7	STUD
025	1 1/8-7 X 6.50	ASTM A193 B7	STUD
026	3/4-10 X 4.50	ASTM A325	BOLT
027	7/8-9 X 4.00	ASTM-A193 B7	STUD
028	3/8-16 X 1.75	ASTM-A193 B7	STUD
029	1 1/4-8 X 10.0	ASTM A193 B7	CAPSCREW
030	1/1/2-8 X 9.25	ASTM A193 B7	STUD