UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-348/92-09 and 50-364/92-09

Licensee: Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

Docket Nos.: 50-348 and 50-364   License Nos.: NPF-2 and NPF-8

Facility name: Farley Units 1 and 2

Inspection Conducted: March 9 - April 13, 1992

Inspectors: For Robert Hoag   5/1/92
George F. Maxwell, Senior Resident Inspector   Date Signed

For Robert Hoag   5/1/92
Michael J. Morgan, Resident Inspector   Date Signed

For Robert Hoag   5/1/92
Robert W. Wright, Project Engineer   Date Signed

Accompanying Personnel: Stepehn T. Hoffman, Project Manager, NRR

Approved by: David M Verrelli   5/1/92
for Floyd S. Cantrell, Chief   Date Signed
Project Section 1B
Division of Reactor Projects

SUMMARY

Scope:

This routine, resident inspection involved inspection in the areas of operations, maintenance activities, surveillance testing, installation and testing of modifications, system walkdown and a continuing evaluation of licensee self-assessment capability. Deep backshift inspections were conducted March 31, April 6 and April 7, 1992.

Results:

A violation was identified during the shutdown for the Unit 2 refueling outage involving previous failure to take appropriate corrective action, paragraph 8.d. On March 7, a Unit 2 pressurizer relief valve lifted inadvertently during performance of a surveillance by mechanical maintenance personnel. Further inspector investigation of this event is on going, paragraph 5.b. On March 19, during performance of a Unit 2 local leak rate test,

about one gallon of contaminated water was spilled on the auxiliary building floor, paragraph 3.a. On March 23, an inadvertent discharge of CO_2 to the "1J" 4160V emergency bus occurred. A follow-up review by the inspectors is ongoing, paragraph 3.b.

New procedures have been implemented to assure the reliability of the decay heat removal system during the current Unit 2 refueling outage. This involves an innovative apprcach, for monitoring the "critical" shutdown safety parameters, paragraph 3.c.

Unit 2, train "B" service water piping was satisfactorily repaired and tested in accordance with ASME Section XI requirements. However, the licensee's seismic II/I hazards and tornado missile risk evaluations for this repair were prepared after work activities were underway, paragraph 4.b. and 6.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*W. Bayne, Supervisor Safety Audit and Engineering Review
R. Coleman, Modification Manager
L. Enfinger, Administrative Manager
*R. Hill, Assistant General Manager - Plant Support
*D. Morey, General Manager - Farley Nuclear Plant
*C. Nesbitt, Operations Manager
J. Osterholtz, Technical Manager
*L. Stinson, Assistant General Manager - Plant Operations
J. Thomas, Maintenance Manager
L. Williams, Training Manager
*J. Woodard, Vice President (Farley) - Southern Nuclear Operating Co.
*B. Yance, Systems Performance Manager

Other licensee employees contacted included, technicians, operations personnel, security, maintenance, I&C and office personnel.

*Attended exit interview

During the week of March 16 to 19, NRR Project Manager, S. Hoffman, met with the resident inspectors and site personnel and conducted an audit of on-going site activities.

During the week of March 16 to 18, Region II Projects Section Chief, F. Cantrell, met with the resident inspectors and site personnel.

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Plant Status

a. Unit 1 Status

Unit 1 operated at approximately 100 percent power for most of the reporting period; however, on April 4, power was reduced to about 85 percent in order to conduct main turbine governor valve and reactor control rod testing/surveillance. Unit power was returned to 100 percent after testing. On April 10, the unit was ramped down to approximately 30 percent power, to allow for investigation and repair of leaking valves associated with the reactor coolant drain tank. The unit was returned to 100 percent power on April 11.

b. Unit 2 Status

Unit 2 continued with scheduled refueling outage number 8 which is expected to continue until approximately May 5, 1992.

c. Other NRC/Licensee Meetings and Inspections

During the week of March 23 - 27, Region II Materials and Processes Section personnel, conducted an inspection of the plant structural supports and bolting inspection practices (Inspection Report 50-348,364/92-08).

During the weeks of March 30 - April 3, and April 13-17, Region II Materials and Processes Section personnel, conducted an audit of licensee S/G tube inspections and the Unit 2 reactor vessel ISI (Inspection Report 50-348,364/92-11).

During the weeks of March 23-27 and March 30 - April 2, Region II Operator Licensing Branch Section personnel, performed a requalification examination of 16 licensed operators. (Inspection Report 50-348/92-300).

During the week of April 6 - April 10, Region II Facilities Radiation Protection Section personnel, conducted an inspection of licensee radiation protection work practices. (Inspection Report 50-348,364/92-10).

3. Operational Safety Verification (71707, 37828, 60710, and TI 2515/113)

The inspectors conducted routine plant tours during this inspection period, in accordance with guidance provided by NRC inspection procedure MC71707 to verify licensee requirements and commitments were being implemented. Inspection tours included review of site documentation, interviews with plant personnel and an on-going evaluation and observation of site security.

The inspectors have noted a reduction in outage related overtime; however, routine use of 12-hour days for both the operating and the outage unit operators is often the rule rather than the exception. Management appears to be keeping close track of work hours. The following reminder was noted in the plant's night order book for March 6, 1992: "People are reminded that during the outage they should keep close track of their work hours. No one is automatically approved to exceed any AP-64 guidelines. Any deviations must receive prior individual ED (emergency director) approval." The inspectors reviewed the circumstances related to the events discussed below to ensure that they understood the event and that it was properly investigated and reported if required:

a. Spill of Contaminated Water During Local Leak Rate Testing - Unit 2

On March 19, at approximately 10:00 p.m., a local leak rate test (LLRT), FNP-2-STP-627, was being performed on reactor building penetration number 28, (penetration associated with the RCP seal water outlet - excess letdown valve numbers Q2E21V249A, V213 and V249B). A vent valve hose associated with this penetration had been routed to an auxiliary building drain; however, the vent valve was left open. During pressurization of this penetration, the hose assembly was rapidly pressurized and caused the hose to become dislodged from the funnel drain. This resulted in the discharge of approximately one gallon of contaminated water to the auxiliary building floor. Some slight clothing contamination was experienced by two individuals involved with the test; however, subsequent monitoring of both individuals did not indicate any skin contamination or intake by anyone.

The spill area was decontaminated and all personnel involved with this and other similar testing were counselled on the need to ensure that vent lines are properly secured. They were also told to ensure that vent valves are closed, when they are required by procedure, and of the need to slowly pressurize or depressurize penetrations undergoing surveillance testing. The actions taken by the licensee for correcting the cause of this event were reviewed by the inspectors and appear to be adequate.

b. Inadvertent Discharge of Fire Protection System Carbon Dioxide (CO_2) Into Bus "1C" - Unit 1

On March 23, at approximately 8:53 a.m., the diesel building fire protection CO_2 automatically discharged into 4160V bus "1J". Upon investigation of the problem by the diesel building system operator and the operations shift foreman, no fire was detected in the area nor the bus. The CO_2 fire protection system to the diesel building was immediately isolated.

Security records indicated that contract personnel (Fluor, MOVATS and Westinghouse) were in the building just prior to actuation but evacuated the area prior to the CO2 discharge. A work request was written to investigate and correct the cause of the discharge. The resident inspectors were not immediately aware of this event and are reviewing the details and reportability of this event. This additional review is being documented as unresolved item (URI 348/92-09-01) pending completion of the staff's assessment on the licensee's reporting of this event.

c. Shutdown Safety Assessment and Reliable Decay Heat Removal During Outages (TI 2515/113)

The inspectors reviewed the following Unit 2 outage procedures and evaluated the reliability of decay heat removal per TI 2515/113:

(1) FNP has issued the following special procedures for use in this outage:

- o FNP-0-SOP-100.0; Shutdown Safety Assessment
- o FNP-2-SOP-1.11; Mid-Loop Operations
- o FNP-2-SOP-14.1; Containment Closure
- o FNP-2-AOP-12.0; Residual Heat Removal Malfunction
- o FNP-2-AOP-5.0; Loss of A or B Train Electrical Power

(2) The following procedures ensure that forced circulation decay heat removal is maintained or if natural circulation is used, all required conditions are met:

- o FNP-0-SOP-100.0; Shutdown Safety Assessment
- o FNP-2-UOP-2.1; Shutdown of Unit from Min Load to HSB (Hot standby)
- o FNP-2-UOP-2.2; Cooldown of the Unit from HSB

(3) The use of SOP-100.0, Shutdown Safety Assessment ensures one offsite power source and one onsite power source is available to each required shutdown load when less than the full complement of power sources is available.

(4) Use of the following procedures ensures required "backup power" is available during outage maintenance:

- o FNP-2-AOP-5.0; Loss of A or B Train Electrical Power
- o FNP-2-AOP 5.1; Contingency Electrical Alignments

During the development of the above "contingency electrical alignment" procedure, an analysis of "non-standard" electrical line-ups and load carrying capacities was performed.

As part of initial operator and requalification training, use of AOP-5.0 and 5.1 is presented in the classroom and performed using the training simulator.

(5) Daily and during each shift a shutdown safety assessment per procedure FNP-0-SOP-100 is made to determine plant vulnerability and available power sources.

(6) FNP declares any of the five D/Gs inoperable whenever field flashing is removed for maintenance or testing.

Plant management has developed the previously mentioned "shutdown safety assessment" tool. It is somewhat similar to the safety status tree configuration in that it results in a Red, Orange, Yellow or Green condition. The determination of condition is made by assigning points to the status of certain shutdown safety system conditions. Upon completion of the assignment of points, the total is calculated and based on this total, individual condition colors are determined for reactivity, core cooling, power availability, containment, inventory, RCS integrity and, if the vessel is in a defueled condition, spent fuel pool cooling. A red condition is prohibited and a green condition is fully acceptable.

This tool has been used throughout this outage and is completed on each shift to ensure awareness of plant safety status. Additionally, the determination of the status is informational. It is intended to keep people informed of where the plant stands with regard to shutdown safety. It is not intended to provide absolute restrictions on plant operations.

These procedures provide a new approach to ensuring availability of fuel cooling and provide management a quick evaluation of plant conditions.

The results of inspections in this overall area indicate the program was effective with respect to meeting the safety objectives. No deviations or violations were identified in this area. One URI was identified to obtain more information on b. above.

4. Monthly Maintenance Observation (62703)

a. The inspectors reviewed various licensee preventative and corrective maintenance activities, in accordance with guidance provided by NRC inspection procedure MC62703, to determine conformance with facility procedures, plant work requests and NRC regulatory requirements.

Portions of the following maintenance activities were observed:

(1) MWR-246240; Replace lagging on "A" charging pump mini-flow

(2) MWR-246832; Replace RCDT flow transmitter indicator

(3) MWR-247248; Investigate and repair B2J sequencer

(4) MWR-248153; Investigate and repair "2B" prime pump seal

(5) MWR-248360; Extraction steam supply to MSR "2A" valve bonnet not installed correctly - reinstall valve bonnet

(6) WO-20900; Replacement of Train "B" Service Water Piping

b. Permanent Repair-Train "B" Service Water Piping - Unit 2

From March 22 through April 13, permanent on-going repairs were performed on Unit 2 train "B" service water piping. The repair served as the replacement of the temporary non-code repair performed in March, 1991 and documented in monthly inspection report 50-348, 364/91-04, paragraph 4. A portion of the piping was replaced with a new section that required three welds to install. The hydrostatic test and the nondestructive tests performed as part of the work package were acceptable in accordance with ASME Section XI requirements.

No deviations or violations were identified in this area. The results of inspections in the maintenance area indicate that both operations and maintenance personnel conducted the above maintenance activities and corresponding tests in accordance with applicable procedures.

5. Monthly Surveillance Observation (61726)

a. The inspectors witnessed surveillance test activities performed on safety-related systems and components, in accordance with guidance contained in NRC inspection procedure MC61726, in order to verify that such activities were performed in accordance with facility procedures and NRC regulatory and Technical Specification requirements.

The following surveillance activities were observed:

1. 1-STP-33.0B; Train "B" Solid State Protection System Test

2. 1-STP-33.1B; Safeguards Test Cabinet Train "B" Functional Test

3. 1-STP-33.2B; Reactor Trip Breaker Train "B" Operability Test

4. 2-STP-40.2; Sequencer Load Shedding Circuit Test

5. 2-STP-80.13; Diesel Generator 2C 1200KW Load Rejection Test

b. Inadvertent Lifting of Pressurizer Relief Valve During Setpoint Testing Activities - Unit 2.

With Unit 2 in a shutdown condition, (Mode 3 - 520 degrees F), for the refueling outage, on March 7, at about 6:45 a.m., pressurizer relief valve Q2B13V031A was inadvertently "lifted" during set-up of test equipment being used for performance of FNP-2-STP-604.0, Pressurizer Code Safety Valve Testing. RCS pressure was being maintained at about 2040 psig for the surveillance test when control room personnel noted a rapid and unexpected decrease in system pressure. All code safety valve and PORV tailpipe temperatures indicated an increase in temperature and the alarm sounded as setpoints were reached. The mechanical maintenance foreman in the control room immediately directed the mechanical maintenance crew in containment to stop the procedure and close the safety valve. RCS pressure stabilized at approximately 1930 psig. Maintenance personnel subsequently reported difficulty in obtaining proper air pressure on the test rig regulator and stopped all valve testing until test rig was repaired.

The shift supervisor talked to the maintenance foreman and emphasized the importance of maintaining total control of test activities. He also noted that if the safety valve had failed to close, an inadvertent safety injection actuation would have occurred at approximately 1850 psig.

Further investigation of this event by the resident inspectors is on-going and will appear in a future report. (IFI 364/92-09-02)

No deviations or violations were identified in this area. With the exception of poor work control by maintenance personnel while testing a pressurizer relief valve the results of inspections indicated that the above surveillance tests were conducted in accordance with applicable procedures.

6. Installation and Testing of Modifications (37700 and 37828)

As part of the continuing review of 10 CFR 50.59 safety evaluations, the NRR Project Manager (PM) conducted an on-site review of the safety evaluation prepared for the repair of a through-wall flaw that was identified in the 12-inch Unit 2, Train B, service water return line from the diesel generator building. On April 22, 1991, the NRC approved an APCO request for temporary relief from the repair requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI (the Code). The relief request permitted a temporary repair of the flaw until the next Unit 2 outage when the permanent Code repair could be performed.

During the review on March 18 and 19, the area adjacent to the PAP had been excavated to allow access to complete the permanent repair of the service water line. This exposed the normally buried service water lines. The PM requested the safety evaluation performed for the repair activities underway. In response to the request, the PM was provided with the following:

- Nuclear Safety Evaluation Checklist for PCN No. S-91-2-7291, dated February 26, 1991, for the temporary repair of the flaw.

- o D. Shelton, Southern Company Services, letter to J. Woodard, Alabama Power Company, "PCR 91-2-7291," dated March 22, 1991.

- o Southern Nuclear Operating Company intracompany correspondence from K. McCracken to B. McKinney, "Farley-Probability of Tornado Missile Damage to Service D/G Return Lines during Excavation," dated March 27, 1991.

- o Farley Nuclear Plant Maintenance Work Request No. 232885 for the repair of the B train service water return from the diesel generators, released for work March 11, 1992.

Upon request PMD personnel provided the documentation listed above and indicated that the activities underway were covered by the evaluations performed for the temporary repair and by existing plant procedures. Documentation that a review was performed to verify that the 10 CFR 50.59 safety evaluation for the temporary repair enveloped the permanent repair activities was not prepared by the licensee. In addition, the following two areas were found where the evaluation of the temporary repair did not fully address current activities:

- o The probabilistic analysis performed to analyze the risk from tornado missiles during the temporary repair assumed that the service water lines were exposed for 16 days. Licensee personnel indicated that the current repair activities required that the lines be exposed for approximately 40 days placing the current activity outside the bounds of the previous analysis.

- o No review was performed for potential hazards from adjacent non-seismically installed components that could potentially fail during a seismic event and damage the exposed service water lines (seismic II/I hazards).

Subsequent to the PM raising the above two issues, the licensee prepared and provided a copy of a March 18, 1992, memorandum from C. Byrd, Southern Company Services, to R. Coleman, APCO PMD, that evaluated the tornado missile risk for the permanent repair activities and found them acceptable. An additional memorandum, dated March 19, 1992, was also prepared and provided a review of the seismic II/I concerns and found them acceptable.

The results of this review indicate that improvement in the safety assessment of non-routine activities and the documentation of these evaluations is warranted.

Adequacy of 10 CFR 50.59 safety evaluations will continue to be reviewed by the NRC during future audits.

7. Engineered Safety System Inspection Unit 2 (71710)

The inspectors and the FNP NRR PM conducted a detailed walkdown of the accessible portions of the spent fuel cooling (SFP) and purification systems for Unit 2. The walkdown was conducted in accordance with guidance provided by NRC inspection procedure MC 71710. The walkdown included comparing the current system line-up procedure to the plant SFP cooling and purification drawing and Section 9.1.3 of the FSAR.

The inspectors looked for equipment conditions which could have degraded the plant performance. Some of those specific conditions which were considered included the correct alignment of cooling system hangers and supports, housekeeping, general condition of SFP valves, system component labeling, instrumentation installation and calibration, positioning of valves, recent modifications and indications at the main control boards in the spent fuel pool and SFP cooling heat exchanger and pump areas.

During the walkdown the inspectors observed the reactor operators in the control room and the system operators which were at assigned plant locations and those assigned to monitor and log SFP cooling pump parameters. Each of the operators were found to be alert and cognizant of the plant system and component operability status.

No violations or deviations were identified.

8. Evaluation of Licensee Self-Assessment Capability (40500)

a. On March 6, the inspectors received, from the plant I&C supervisor, summary information of a February 12, 1992 rod control system meeting between representatives of FNP, Southern Nuclear Operating Company and Westinghouse. The evaluation of rod control circuitry card failures, (as noted in Unit 2 LER 91-01 and inspection report 50-348,364/91-06), continues.

b. On March 6, the inspectors evaluated a problem report associated with pressurizer pressure transmitter P-444, (See inspection report 50-348,364/91-20, paragraph 4.d.) The transmitter card was found to be spiking intermittently. It was replaced and the defective card sent to Westinghouse for analysis. In general, the main problem appears to be isolated to the card power supply section transistors and the AC voltage regulators.

c. On March 6, the inspectors evaluated a problem report associated with inaccurate instrumentation drawings associated with Unit 1 over-temperature delta-temperature instruments (See Unit 1 LER 91-06 and inspection report 50-348,364/91-12, paragraph 2.b.(1)). All similar drawings (7300 drawings) were reviewed for their accuracy, in response to the above LER. Drawings which were not correct were revised to correctly reflect the "as-wired" conditions and updates have been forwarded to the facility document control group.

d. Intermediate Range High Flux Reactor Trip And Inadequate Corrective Action - Unit 2

On March 6, 1992, during a plant shutdown for a planned Unit 2 refueling outage, a preventable, automatic reactor trip occurred. This trip was caused by change in the reactor nuclear flux distribution characteristics and related "high neutron flux" effects on the intermediate range (IR) nuclear instrumentation and trip circuitry. When power fell below the P-10 reset point, the IR high flux signals were unblocked and because reactor power level was actually above the high flux trip set point and the high flux bi-stables had not been reset, the reactor tripped. This event was reported to the NRC and LER 92-02 was issued. However, similar "high neutron flux" effects on the IR instruments resulted in a February, 1984, Unit 1 reactor trip. The available intermediate range instrument was bypassed in order to conduct a planned shutdown of the Unit 2 reactor in October, 1990.

On October 13, 1990, during a scheduled Unit 2 reactor shutdown for refueling, Technical Specification 3.0.3 was voluntarily entered. This was necessary because both intermediate range instruments were inoperable; one due to loss of cover gas and the other due to bypassing in order to preclude a reactor trip which could have occurred due to the above redistribution characteristics. This event was documented in Unit 2 LER 90-03 and FNP incident report IR 2-90-310. The permanent corrective action prescribed by IR 2-90-310 was to revise the plant shutdown procedure and to change the nominal reset points for the intermediate range instruments and trip circuitry.

On February 10, 1984, Unit 1 tripped at approximately 10 percent power due to an Intermediate Range High Flux trip signal. Unit 1 LER 84-02 noted that this trip was also caused by change in the reactor nuclear flux distribution characteristics and related "high neutron flux" effects. As stated in the LER; "...when reactor power decreased below the P-10 reset point, the Intermediate Range High Flux signals were automatically unblocked and, since the power level was still above the reset point for the Intermediate Range High Flux bi-stables, the reactor tripped".

If the suggested corrective actions had been implemented in a timely manner for the February, 1984 and the October, 1990 events, this recent challenge to the Unit 2 reactor protection system and the resulting trip could have been prevented. Lack of timeliness and an inadequacy in the performance of proposed corrective actions failed to prevent a Unit 2 reactor trip and is a violation of 10 CFR 50, Appendix B, Criterion XVI(50-364/92-09-01).

No other violations or deviations were identified in this area. The inspectors observed concerted efforts by management to resolve key safety issues.

9. Exit Interview

The inspection scope and findings were summarized during management interviews throughout the report period, and on April 13, with the plant manager and selected members of his staff. The inspection findings were discussed in detail. The licensee acknowledged the inspection findings and did not identify as proprietary any material reviewed by the inspectors during this inspection.

ITEM NUMBER	DESCRIPTION AND REFERENCE
364/92-09-01 (NOV)	Lack of timeliness and an inadequacy in the performance of proposed corrective actions failed to prevent a Unit 2 reactor trip
348/92-09-01 (URI)	Discharge of CO_2 system in D/G building
364/92-09-02 (IFI)	Unexpected/Inadvertent lifting of Unit 2 pressurizer relief valve

10. Acronyms and Abbreviations

AFW	-	Auxiliary Feedwater
ALARA	-	"As Low As Reasonably Achievable"
AOP	-	Abnormal Operating Procedure
AP	-	Administrative Procedure
APCO	-	Alabama Power Company
BOP	-	Balance of Plant
BTRS	-	Boron Thermal Regeneration System
CFR	-	Code of Federal Regulations
CVCS	-	Chemical and Volume Control System
CCW	-	Component Cooling Water
CSTS	-	Condensate Storage Tank System
CS	-	Containment Spray System
DDFP	-	Diesel Driven Fire Pump
D/G	-	Emergency Diesel Generator
DRP	-	Division of Reactor Projects
DPM	-	Disintegration Per Minute
ECP	-	Emergency Contingency Procedure
EIP	-	Emergency Plant Implementing Procedure
EPA	-	Environmental Protection Agency
EQ	-	Environmental Qualifications
ESF	-	Engineered Safety Features
EWR	-	Engineering Work Request
F	-	Fahrenheit
FNP	-	Farley Nuclear Plant
FSP	-	Fire Surveillance Procedure
GPM	-	Gallons Per Minute
ISI	-	Inservice Inspection
IST	-	Inservice Test
LCO	-	Limiting Condition for Operation
MDFP	-	Motor Driven Fire Pump
MESG	-	Maintenance and Engineering Support Group
MOV	-	Motor-Operated Valve
MOVATS	-	Motor-Operated Valve Actuation Testing
MWR	-	Maintenance Work Request
NCR	-	Nonconformance Report
NRC	-	Nuclear Regulatory Commission
NRR	-	NR Office of Nuclear Reactor Regulation
OATC	-	Operator at the Controls
PAP	-	Primary Access Point
PCCV	-	Positive Closing Check Valve
PCN	-	Plant Change Notice
PCR	-	Plant Change Request
PMD	-	Plant Modifications Department
PORV	-	Power Operated Relief Valve
PPB	-	Parts Per Billion
PPM	-	Parts Per Million
PRT	-	Pressurizer Relief Tank
PSID	-	Pressure per Square Inch Differential
PVC	-	Polyvinyl Chloride
PZR	-	Pressurizer

RCDT	-	Reactor Coolant Drain Tank
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RTD	-	Resistance Temperature Detector
SI	-	Safety Injection
S/G	-	Steam Generator
SAER	-	Safety Audit and Engineering Review
SFO	-	Shift Foreman - Operating
SGFP	-	Steam Generator Feedwater Pump
SO	-	Systems Operator
SFP	-	Spent Fuel Pool
SOP	-	Standard Operation Procedure
SPDS	-	Safety Parameter Display System
SS	-	Shift Supervisor
SSPS	-	Solid State Protection System
STP	-	Surveillance Test Procedure
SWS	-	Service Water System
TS	-	Technical Specification
TSC	-	Technical Support Center
VDC	-	Voltage Direct Current
WA	-	Work Authorization