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

Report Nos.: 50-250/87-27 and 50-251/87-27

Licensee: Florida Power and Light Company
9250 West Flagler Street
Miami, FL 33102

Docket Nos.: 50-250 and 50-251 License Nos.: DPR-31 and DPR-41

Facility Name: Turkey Point 3 and 4

Inspection Conducted: May 18 - June 22, 1987

Inspectors:	D. R. Brewer, Senior Resident Inspector	July 14, 87 Date Signed
	K. W. Van Dyne, Resident Inspector	July 14, 1987 Date Signed
	J. B. Macdonald, Resident Inspector	July 14, 87 Date Signed
Approved by:	Bruce Wilson, Section Chief Division of Reactor Projects	7/15/87 Date Signed

SUMMARY

Scope: This routine, unannounced inspection entailed direct inspection at the site, including backshift inspection, in the areas of annual and monthly surveillance, maintenance observations and reviews, operational safety, and plant events.

Results: Violation - Failure to meet the requirements of Technical Specification (TS) 4.1, Table 4.1-1, Sheet 4, Channel Description Item 38.b (paragraph 7).

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   C. M. Wethy, Vice President - Turkey Point
   *C. J. Baker, Plant Manager-Nuclear - Turkey Point
   *F. H. Southworth, Maintenance Superintendent - Nuclear
   D. A. Chaney, Site Engineering Manager (SEM)
   *D. D. Grandage, Operations Superintendent and Acting Plant Manager
   T. A. Finn, Training Supervisor
   J. Webb, Operations - Maintenance Coordinator
   J. W. Kappes, Performance Enhancement Coordinator
   R. A. Longtemps, Mechanical Maintenance Department Supervisor
   D. Tomasewski, Instrument and Control (IC) Department Supervisor
   J. C. Strong, Electrical Department Supervisor
   *W. Bladow, Quality Assurance (QA) Superintendent
   M. J. Crisler, Quality Control (QC) Supervisor
   *J. A. Labarraque, Technical Department Supervisor
   V. Kaminskas, Reactor Engineering Supervisor
   R. G. Mende, Operations Supervisor
   *J. Arias, Regulation and Compliance Supervisor
   *R. Hart, Regulation and Compliance Engineer
   *W. C. Miller, Senior Technical Advisor
   P. W. Hughes, Health Physics Supervisor
   G. Solomon, Regulation and Compliance Engineer
   *J. Donis, Engineering Department Supervisor
   *E. F. Hayes, Quality Control Supervisor

   Other licensee employees contacted included construction craftsmen, engineers, technicians, operators, mechanics, and electricians.

   *Attended exit interview on June 22, 1987.

2. Exit Interview

   The inspection scope and findings were summarized during management interviews held throughout the reporting period with the Plant Manager-Nuclear and selected members of his staff. An exit meeting was conducted on June 22, 1987. The areas requiring management attention were reviewed. The licensee acknowledged the findings without exception.

   One violation was identified:

   Failure to meet the requirements of TS 4.1, Table 4.1-1 Sheet 4, Channel Description Item 38.b, in that, surveillance testing of the Unit 3 spent fuel exhaust monitor had not been performed within its required periodicity (paragraph 7) (250,251/87-27-01).

One Unresolved Item (URI) was identified.

NRC review of Intake Cooling Water (ICW) system operation outside its design basis. (Paragraph 9) (URI 250,251/87-27-02).

3. Followup on Items of Noncompliance (92702)

A review was conducted of the following noncompliances to assure that corrective actions were adequately implemented and resulted in conformance with regulatory requirements. Verification of corrective action was achieved through record reviews, observation and discussions with licensee personnel. Licensee correspondence was evaluated to ensure that the responses were timely and that corrective actions were implemented within the time periods specified in the reply.

(Closed) Violation 251/83-39-02, Failure to complete Post-Trip review prior to restart. The licensee determined that because the trip occurred from a subcritical condition a post-trip review was not required. Administrative Procedure (AP) 0102.16 currently requires that a post-trip review be performed for any automatic or required manual reactor trip or turbine trip including subcritical reactor trips. This item is closed.

(Closed) Violation 250,251/84-04-01, Auxiliary Feedwater (AFW) Pump Operability. From December 5, 1983 to December 16, 1983, several undocumented AFW pump runs were performed. During these runs the governor manual speed control knob was positioned to the minimum speed setting on the A and C AFW pumps. Subsequently, when the next required TS surveillance test was performed the A and C AFW pumps failed to attain rated speed. The pumps were therefore considered out of service since December 17, 1983 and the 72 hour TS Limiting Condition for Operations (LCO) for AFW pump operability was exceeded. The licensee has implemented corrective actions to preclude recurrence of this violation. The knob position is checked once per shift by the Nuclear Turbine Operator and logged in 3-OSP-201.3, NPO Nuclear Plant Operator Daily Logs. The knob position is independently verified following operation in 3-OSP-075.1 (2) AFW Train 1 (2) Operability Verification and testing in 3-OSP-075.6 (7) AFW Train 1 (2) Operability Verification. The corrective actions were reviewed and determined to be adequate. Violation 250,251/84-04-01 is closed.

(Closed) Violation 251/83-38-03, T.S. 6.8.1, Failure to implement and follow procedures. This violation was issued as a result of five examples of non-adherence to existing procedures. Example one included annunciator and indicator discrepancies in the control room that were not addressed. Operations personnel were instructed to monitor all equipment and to submit Plant Work Orders (PWO's) for malfunctioning equipment. Example two involved 2 licensee personnel who failed to properly sign out of a locked high radiation area. AP 11550.101, Radiological Investigation Reports, was revised to implement incident reporting techniques and the individuals involved were restricted from the Radiation Control Area (RCA). Example three was a case in which the results of a local leak rate test were not available. OP-13514.2, Containment Access Hatch Local Leak

Rate Test, was upgraded to clarify the intended requirement of the test and the systems performance group personnel were instructed of the need to document all discrepancies identified during surveillance testing. Example four was a case in which on October 2, 1983, an oncoming operator did not perform an adequate review of the log sheet. AP 0103.2, Responsibilities of Operators and Shift Technicians On Shift and Maintenance of Operating Logs and Records, has been revised to include watch relief checklists. The event above is considered an isolated example, in that the oncoming operator was an emergency relief and was therefore not part of the normal shift turnover process. Example five was a case in which on October 2, 1983, a nuclear operator failed to tag valves he had isolated thereby preventing later independent verification of the valves. AP 0103.31, Independent Verification (IV), was revised to strengthen control of the IV program. The procedure more clearly defines IV requirements on safety related systems and emphasizes the importance of a strong IV program. The corrective actions were reviewed and determined to be adequate. Violation 251/83-38-03 is closed.

4. Followup on URIs, Inspector Followup Item (IFIs), Inspection & Enforcement Information Notices (IENs), IE Bulletins (IEBs) (information only), NRC Requests (92701)

A review was conducted of the following items to assure that adequate applicability reviews were performed, appropriate distribution was made, and if required, adequate and timely corrective actions were taken or planned.

(Closed) IFI 250,251/84-09-03, Inadequate On-The Spot Changes (OTSCs) to EP 20007 and OP 7309.1. The specific concerns which are collectively tracked via this IFI have been resolved as indicated by the review of AP 0109.3, revision dated 8/12/86.

(Closed) IFI 250/84-23-07, 251/84-24-07, Revising In-Sevice Test (IST) Program. On October 11, 1985, a change to the April 2, 1984 In-service Inspection (ISI) and IST TS was requested. Additionally, surveillance procedures have been written to provide guidelines for IST testing of the spent fuel pit pumps, the charging pumps and the boric acid transfer pumps.

(Closed) IFI 250/84-34-07, 251/84-35-07, Health Physics (HP) discrepancies in Radiation Work Permit (RWP) requirements. OP 11550.2 has been superseded by O-HPA-002 which requires adherence by all personnel to the RWP. Additionally, RWP 87-02, which applies to all HP personnel, requires that dress out requirements as posted for contaminated areas be followed.

(Closed) IFI 250/84-34-10, 251/84-35-10, Graphitic corrosion of heat exchangers. In 1981, the licensee replaced the cast iron channels and heads of all Component Cooling Water (CCW) heat exchangers with those made of carbon steel. Carbon steel does not exhibit the same corrosion problems identified in IE Notice 84-71. In addition, the Unit 3 CCW heat exchanger channels and heads were replaced during the current refueling outage with those made of monel. Similar replacement is scheduled for Unit 4 during its next refueling outage.

(Closed) IFI 250/84-34-08, 251/84-35-08, Reactor Protection System (RPS) Pushbutton Test. While performing OP-1004.2 RPS-Periodic Test unexpected trip signals were received. Instrument and Control personnel reviewed the logic being tested and discovered that the signals were valid and appropriate for the testing sequence being performed if the pushbuttons were not depressed simultaneously. The procedure was subsequently cancelled and replaced by unit specific procedures 3/4-OSP-049.1 RPS Logic Test. Communications during testing was also upgraded by the use of headsets rather than radios. IFI 250/84-34-08, 251/84-35-08 is closed.

(Closed) IFI 250/84-35-07,251/84-36-07, Surveillance of Emergency Diesel Generator (EDG) Skid Tank Level Switches. This IFI was written to ensure that the EDG skid tank level switches were tested in a controlled surveillance procedure. The switches are calibrated per 0-PMI_023.1, EDG Instrument Calibration. IFI 250/84-35-07, 251/84-36-07 is closed.

5. Unresolved Items (URI)

An URI is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation. One URI is addressed in paragraph 9 (URI 250,251/87-27-02).

6. Onsite Followup and In-Office Review of Written Reports Of Nonroutine Events (92700 and 90712)

The Licensee Event Reports (LER) discussed below were reviewed and closed. The Inspectors verified that reporting requirements had been met, root cause analysis was performed, corrective actions appeared appropriate, and generic applicability had been considered. Additionally, the Inspectors verified that the licensee had reviewed the event, corrective actions were implemented, responsibility for corrective actions not fully completed was clearly assigned, safety questions had been evaluated and resolved, and violations of regulations or TS conditions had been identified.

(Closed) LER 250/85-36, Residual Heat Removal Capability. This event describes how the failure of a relay (PC-403-A-2) caused residual heat removal (RHR) system suction valve (MOV-3-750) to close which resulted in loss of RHR capability. The inspectors verified corrective action per the LER (by plant work orders 63-8089 and 8084) was performed. This item is closed.

(Closed) LER 250/86-014, Technical Specification Emergency Diesel Generator. The 'B' EDG failed to achieve the required 200 rpm in less than 15 seconds while being tested in accordance with TS 3.7.2.b. Both units were in cold shutdown during the time of the event. The inspector reviewed the corrective actions taken by the licensee, which did not result in the identification of an apparent root cause. The documented review performed by the licensee appeared complete and subsequent restarts were successfully performed. This item is closed.

(Closed) LER 250/86-26, Turbine Stop Valve Failure to Close. On June 12, 1986, a Unit 3 test of high pressure turbine valve operability revealed that the left high pressure stop valve would not close completely. In addition, the south west intercept and reheat stop valves failed to close. Load was subsequently reduced to less than 40 percent as required by Technical Specification.. Investigation revealed that the turbine stop valve pilot had malfunctioned and the intercept and reheat stop valve control circuit leads were incorrectly landed. Following repairs, valve operability was verified by further testing and the unit was returned to 100 percent power. This item is closed.

(Closed) LER 250/86-12, Reactor Protection System Actuation-Subcritical Trip. On March 5, 1986, while in hot standby, the Unit 3 reactor tripped while subcritical. The cause of the trip was a failure of the detector and first stage preamplifier of Source Range Detector N-31. The failed parts were replaced and N-31 passed the functional test. The post trip review did not identify any significant problems. This item is closed.

(Closed) LER 251/84-17, Engineered Safety Feature Actuation-Reactor Trip. This event was caused by an incorrect switching order originating from the fossil units 1 and 2 control room. This item is closed based on the following referenced corrective action: 1) R. L. Taylor to W. E. Coe letter, dated 8/8/84 2) C. M. Mennes to O. R. Whitney letter, dated 6/30/86 and 3) Design Verification Reports and System Acceptance/Turnover Sheets for PC/Ms 84-137 and 84-138.

(Closed) LER 250/84-25, Intake Cooling Water System. This LER was generated as a result of concerns regarding operability and design basis of the ICW/CCW systems. Violation 250/84-29-01 and 251/84-30-02 was subsequently issued. Similar concerns have been addressed in this report and are being tracked via URI 250,251/87-27-02. The Justification for continued oprations (JCO)/Safety Evaluation JPE-L-85-38 is currently being reviewed by NRR for technical adequacy and regulatory compliance. LER 250/84-25 is closed.

The following LERs were reviewed and closed based on an in-office review. The inspectors verified that reporting requirements had been met, root cause analysis was performed, corrective actions appeared appropriate, and generic applicability had been considered. In addition, each LER was reviewed for and determined not to require further onsite inspector followup.

LER 250/86-17	LER 250/86-15
LER 251/86-61	LER 251/84-02
LER 251/86-07	LER 251/84-24
LER 250/84-27	

7. Monthly and Annual Surveillance Observation (61726/61700)

The inspectors observed TS required surveillance testing and verified: that the test procedure conformed to the requirements of the TS, that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation (LCO) were met, that test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test, that deficiencies were identified, as appropriate, and were properly reviewed and resolved by management personnel and that system restoration was adequate. For completed tests, the inspectors verified that testing frequencies were met and tests were performed by qualified individuals.

The inspectors witnessed/reviewed portions of the following test activities:

Dual Unit Loss of Offsite Power With Single Unit Safety Injection, Test Procedure (TP)-336 (Units 3 and 4). Details of this testing are contained in report 250,251/87-26.

Spent fuel pit exhaust monitor monthly surveillance

On May 14, 1987, the licensee discovered that the monthly TS required surveillance of the spent fuel pit exhaust monitor had not been performed since March 11, 1987, TS 4.1, Table 4.1-1, Sheet 4, Item 38b requires that the monitor to be tested monthly. The test was required to be performed prior to April 18, 1987. The surveillance test was performed with acceptable results on May 15, 1987. The cause of the missed surveillance appears to have been a misinterpretation by operations personnel of the requirements of O-OSP-200.1, Schedule of Plant Checks and Surveillances. The surveillance test procedure O-PMI-067.11 Process Radiation Monitoring System (PRMS) SPING Functional Test Procedure, was scheduled, by O-OSP-200.1, to be performed on April 8, 1987. The due date was April 11, 1987 and its late date was April 18, 1987. Quality Control (QC) personnel alerted the Instrumentation and Controls department via memo on April 6 and 13 that the surveillance was required by April 18. On April 18, QC notified the GEMS coordinator of the surveillance requirement. The GEMS coordinator brought the concern to the Plant Supervisor-Nuclear (PSN) on shift. The PSN reviewed O-OSP-200.1, page 33 and misinterpreted a foot-note to mean that the surveillance procedure was not required in the applicable mode of operation. The footnote in fact meant that only the portion of the surveillance procedure testing the condenser air ejector and main steam line monitors were not required, the spent fuel pit exhaust monitor was still required by TS to be surveillance tested. The failure to perform a TS required surveillance is a violation (250,251/87-27-01).

8. Maintenance Observations (62703/62700/37701)

Station maintenance activities of safety related systems and components were observed and reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards and in conformance with TS.

The following items were considered during this review, as appropriate: that LCOs were met while components or systems were removed from service; that approvals were obtained prior to initiating work; that activities were accomplished using approved procedures and were inspected as applicable; that procedures used were adequate to control the activity; that troubleshooting activities were controlled and repair records accurately reflected the maintenance performed; that functional testing and/or calibrations were performed prior to returning components or systems to service; that QC records were maintained; that activities were accomplished by qualified personnel; that parts and materials used were properly certified; that radiological controls were properly implemented; that QC hold points were established and observed where required; that fire prevention controls were implemented; that outside contractor force activities were controlled in accordance with the approved QA program; and that housekeeping was actively pursued.

The following maintenance activities were observed and/or reviewed:

Repair of the safeguards system selector switch (PWO 63-6176).

Repair of boric acid transfer pump seals (PWOs 63-2334, 2437, and 2661).

A. During this inspection period the inspectors reviewed the circumstances involving the identification and disposition of an apparent discrepancy in the Unit 3 containment floor (elevation 14'0"). While drilling 3/4" holes in the containment floor for the installation of three Hilti bolts, construction workers observed a "void" in the concrete. When the drill was removed, water was present at the bottom of the hole. Health Physics personnel surveyed the material deposited from the drill at 40,000 decays per minute (dpm). Discrepant field condition report (DFC)-228-87 and non conformance report (NCR)-C-513-87 were written to identify and resolve the discrepancy.

Subsequent QC inspections, engineering review/evaluation and final resolution were determined to be adequate. Material contamination levels were discussed with Health Physics management personnel and were determined to be within the expected range. The identified "void" was actually compressible expansion joint material between the floor slab and the containment liner. No damage to the liner plate was determined to have occurred from drilling and no significant accumulation of water was indicated between the liner plate and the slab. Engineering review determined, therefore, that no containment operability concern exists and specified that the holes be repaired by grouting.

B. On June 4, 1987, the licensee issued a safety evaluation and justification for continued operation (JCO) in the event of loss of Heating, Ventilation and Air Condition (HVAC) to DC equipment and inverter rooms (JPE-LR-87-20, Rev. 1). This JCO will be utilized until planned modifications to the DC equipment/inverter rooms HVAC system are implemented. The loss of HVAC in the DC equipment/inverter rooms

could result in unacceptably high temperatures, without operator action. Therefore, detectability of high temperatures in the DC equipment/inverter rooms is essential. Because there are no formal procedures in place which require temperature surveillance of the rooms the loss of HVAC in the DC equipment/inverter rooms involves an unanalyzed condition. The licensee maintains that a substantial safety hazard does not exist due to the low probability of room temperature excursion proceeding to the point where the nuclear safety related consequences would be significant. The use of supplemental cooling (portable fans) can be used to reduce and to maintain room temperatures within electrical equipment design temperatures and thus provide additional assurance that electrical equipment will not fail due to high room temperatures. Operation should continue with fire doors open such that the EDG-backed A/C unit can provide cooling to the north and south areas of the Control Building Annex. Implementation of appropriate plant changes eliminate the concern that resulted in an unanalyzed condition determination. That is, operator action must be relied upon to detect and terminate loss of HVAC induced temperature excursions. This JCO is under NRC review.

No violations or deviations were identified within the areas inspected.

9. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, conducted discussions with control room operators, observed shift turnovers and confirmed operability of instrumentation. The inspectors verified the operability of selected emergency systems, verified that maintenance work orders had been submitted as required and that followup and prioritization of work was accomplished. The inspectors reviewed tagout records, verified compliance with TS LCOs and verified the return to service of affected components.

By observation and direct interviews, verification was made that the physical security plan was being implemented.

Plant housekeeping/cleanliness conditions and implementation of radiological controls were observed.

Tours of the intake structure and diesel, auxiliary, control and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks and excessive vibrations.

The inspectors walked down accessible portions of the following safety related systems to verify operability and proper valve/switch alignment:

A and B Emergency Diesel Generators
Control Room Vertical Panels and Safeguards Racks
Spent Fuel Pool (Units 3 and 4)
4160 Volt Load and 480 Volt Motor Control Centers (Units 3 and 4)
Boric Acid Transfer Pump Nitrogen Seal System (Units 3 and 4)

A. As a result of the design basis reconstitution effort, two engineering safety evaluation packages were issued during this report period which affect emergency safety features (ESF) equipment.

1. Westinghouse safety evaluation (SECL-87-223, Rev.2) as transmitted in letter JPES-PTP-87-1037 describes a design deficiency in the containment spray system. A review of the Westinghouse system calculation from 1971 illustrates that credit was taken for a restricting orifice that is not installed in the piping. The safety evaluation specifies plant change/modifications to correct this deficiency. Licensee Event Report 250/87-14 will be utilized to ensure proper implementation of corrective action.

2. Licensee safety evaluation JPE-LR-87-017 describes a recent Westinghouse analysis that indicates the original Emergency Core Cooling System (ECCS) switchover design and procedure could not be demonstrated to be acceptable using current analysis techniques and assumptions. These analyses show that no interruption in ECCS flow greater than about two minutes can be tolerated without potentially exceeding peak clad temperature requirements for a large break loss of coolant accident.

   In order to resolve this concern a permanent change in Emergency Operating Procedures (EOPs) (ES-1.3 and ES-1.4) are required prior to either Turkey Point Nuclear Unit entering operating mode 4. The safety evaluation concludes that the proposed changes to the two EOPs may be implemented without prior NRC approval under the provisions of 10 CFR 50.59.

B. On May 28, 1987, while sequencer testing was in progress, both the normal and emergency boric acid flow paths to the core were lost. An attempt was made to establish a flow path from the refueling water storage tank (RWST) through the charging pumps. The first attempt was unsuccessful, however, after venting the charging pump and the RWST line to the charging pumps, a flow path was verified. After approximately 1 hour the 3B boric acid pump was aligned to unit 4 to insure a flow path while the charging pump was being vented.

Two hours later, while refueling operations were in progress on Unit 3, an unsuccessful attempt was made to borate the Unit 3 core. Core alterations were stopped and the 3B boric acid pump was isolated for maintenance. A flow path was established to Unit 3 within 2 hours and twenty minutes. The loss of a boric acid flow path to Unit 3 was reported to the NRC as an unusual event since no alternate flow path was available with the RWST drained for refueling. At this time, it was thought that a loss of suction occurred to the 3B boric acid pump because it was aligned to C Boric Acid Storage Tank (BAST). PWO's were written to mechanical maintenance and electrical department to check heat tracing circuits and possible line blockage on the boric acid tank outlet line.

An emergency response team (ERT) was established on June 4, 1987 to review the circumstances surrounding the loss of boric acid flow paths. The purpose of the ERT was to determine root cause and to develop corrective action to prevent similar occurrences.

This event was the subject of an NRC team inspection during the week of June 15 - 19 and details of the team findings are in report 250,251/87-28.

C. ICW/CCW System

Review of Technical Specification Limiting Condition for Operation (LCO) entries for out of service Component Cooling Water (CCW) heat exchangers indicates that repetitive entries into LCOs were required for cleaning CCW heat exchangers throughout 1986. The operation of the CCW system and the Intake Cooling Water (ICW) system, as well, are discussed in a licensee safety evaluation (JPE-L-85-38).

On February 13, 1986 the licensee determined that the ICW system contained two valves which were susceptible to single active failures. The discrepancies were evaluated as not reportable under 10 CFR 21 in Substantial Safety Hazards Evaluation JPE-L-85-38, Revision 0. The licensee determined that the inability to accommodate a single failure constituted an unnecessary contribution to overall risk. Consequently, plans were implemented to evaluate and modify the ICW system to correct the condition.

It was subsequently determined that the inability for one of the valves to perform its function was inconsequential provided that certain parameters were maintained. Revision 1 to JPE-L-85-38 was issued on February 16, 1986 to promulgate graphs depicting the relationship of post accident ICW flow through the CCW heat exchangers, ICW system (cooling canal) temperature, and CCW heat exchanger cleanliness. Based on these parameters, the licensee was able to determine when personnel were to be stationed at a manual isolation valve to shut the valve subsequent to the occurrence of the Maximum Hypothetical Accident (MHA). The corrective actions specified in Revision 0 were expanded to include the development of a CCW heat exchanger performance monitoring program to ensure that the heat transfer capability of the heat exchangers remained sufficient for effective accident mitigation.

It was determined that the effectiveness of the heat exchangers was heavily dependent on precipitation of calcium carbonate from the canal water on the heat exchanger tubes. The high levels of calcium carbonate in the canal system rapidly degraded the heat transfer capability of the heat exchangers. Consequently, the licensee implemented a program to periodically clean them. Cleaning was required approximately weekly, during the summer months, based on the graphs contained in Revision 1. The licensee erroneously assumed

that each cleaning effort returned the heat exchanger performance to its optimum efficiency as calculated by baseline data. Baseline efficiency curves were developed from performance testing data obtained after thorough heat exchanger cleaning by rodding.

In June 1986, it was postulated that, with one heat exchanger out of service for cleaning, canal temperatures might rise to a point where the remaining two heat exchangers could not handle the MHA heat load even after posting an operator at the manual valve. Revision 2 to JPE-L-85-38 was issued on August 5, 1986 to address this possibility.

Revision 2 states;

Should, during the 24 hour LCO period for the cleaning of a CCW heat exchanger, the performance of the remaining two heat exchangers degrade to the point where two ICW pumps flow is necessary to remove the accident heat load the plant shall be placed into the hot shutdown condition within 6 hours unless: (1) It is verified that all 3 ICW pumps are avaiable for service....; <u>and</u>, (2) a total of 24 cumulative hours per three months of operation in this condition have not accrued.

Operation of the Units for 24 hours in a degraded condition such that the flow of two ICW pumps was required to provide accident protection differs with system capability discussions found in the Final Safety Analysis Report and the Technical Specification Bases.

The FSAR, Section 9.3 states, following a loss of coolant accident, two CCW heat exchangers accommodate the heat removal loads. If a CCW heat exchanger fails, the standby heat exchanger provides a 50 percent backup. Additionally, FSAR Table 9.3-5 specifies that two CCW heat exchangers can carry the total emergency heat load. The FSAR specifies, in Section 9.6, that only one ICW pump is required following a MHA and that the minimum operating requirements for the ICW system are met by one pump and one loop header.

The TS Bases specify, in Section B3.4.4, that one CCW pump and two CCW heat exchangers meet the requirements of the MHA analysis.

Section B3.4.5 specifies that one ICW pump meets the requirements of the MHA analysis.

A particular concern of the inspectors, was the operational condition of the Unit 3 CCW heat exchangers in December 1986. On December 1, 1986, a performance test conducted on the Unit 3 CCW heat exchangers, indicated degraded performance. The actual R value, a non dimensional resistance to heat transfer coefficient, calculated from the test data was much greater than the predicted R value of the baseline data. The probable cause of the high R values was inadequate cleaning efforts that failed to remove the heat transfer inhibiting calcium carbonate. The maximum allowable ICW inlet temperature which is the limiting factor for continued operation in the various ICW pump, CCW heat exchanger, and manual operator action scenarios, is directly impacted by the R value. The greater the R value the lower the maximum allowable ICW inlet temperature would be. The R value calculated from the Unit 3 performance test of December 1, 1986 lowered the performance curves such that ICW maximum allowable inlet temperature was reduced by the approximately 10 degrees F. (same slope but the y-intercept was reduced by 10 degrees F.). The revised data and proposed immediate cleaning schedule were forwarded to the Shift Technical Advisors (STAs) on December 4, 1986. The revised data was not implemented and the cleaning schedule was not adhered to. This resulted in the Unit 4 heat exchangers being cleaned prior to the Unit 3 heat exchanger. The Unit 3 cleaning process was not initiated until December 11, 1986, which was contrary to the sequence prescribed in the December 4, 1986 letter. The most degraded heat exchanger (3A CCW heat exchanger) was not cleaned first, further increasing the R value. On June 12, 1987, inspectors brought to the attention of plant management that Unit 3 appeared to have operated outside the design basis of the CCW system for significant periods of time from December 1-12, 1986. The inspectors were particularly concerned about the seventeen hour period (4:45 A.M.-9:45 P.M.) on December 11, 1986 that the 3B CCW heat exchanger was out of service for cleaning. During this seventeen hour period, the two CCW heat exchangers in service would not have been able to dissipate MHA heat load even with the ICW flow rate provided by two ICW pumps as described in safety evaluation JPE-L-85-38, rev. 2, and the turbine plant cooling water system isolated.

On June 16, 1987, Plant Management acknowledged that Unit 3 operated outside the CCW system design basis for the time period in question and the appropriate 10 CFR 50.72 report was made.

Pending NRC staff review of the concern that the licensee has authorized, via a safety evaluation, operation of the ICW system outside its design basis, this issue will be identified as an unresolved item (URI 250,251/87-27-02).

No violations or deviations were identified in the areas inspected.

10. Plant Events (93702)

The following plant events were reviewed to determine facility status and the need for further followup action. Plant parameters were evaluated during transient response. The significance of the event was evaluated along with the performance of the appropriate safety systems and the actions taken by the licensee. The inspectors verified that required notifications were made to the NRC. Evaluations were performed relative to the need for additional NRC response to the event. Additionally, the following issues were examined, as appropriate: details regarding the cause of the event; event chronology; safety system performance; licensee compliance with approved procedures; radiological consequences, if any; and proposed corrective actions. The licensee plans to issue LERs on each event within 30 days following the date of occurrence.

On May 19, 1987, while Unit 3 was in refueling shutdown and Unit 4 in cold shutdown, a discrepancy in the containment spray system was identified. A flow restricting orifice was not installed as assumed in design calculations. This subject was discussed earlier in paragraph 9.

On May 22, 1987, while Unit 3 was in refueling shutdown and Unit 4 in cold shutdown, work in the 4B sequencer cabinet resulted in a "false" undervoltage signal in the sequencer logic. Bus stripping occurred and B EDG automatically started. Operations personnel restored power to the 4B 4160 volt Bus and shutdown the B EDG.

On May 22, 1987, while Unit 3 was in refueling shutdown and Unit 4 in cold shutdown, an issue concerning the inverter room ventilation was identified as a result of the design basis reconstitution effort. In the event of the loss of offsite power, the failure of an air condition unit would result in the loss of ventilation to the inverter rooms. This subject was addressed earlier in paragraph 8.

On May 26, 1987, while Unit 3 was in refueling shutdown, a partial train engineering safeguards features (ESF) actuation occurred while reenergizing safeguards rack 45 in preparation for the performance of TP-336, Dual Unit Loss of Offsite Power with Single Unit Safety Injection. The cause was attributed to the apparent malfunctioning of the safeguards selector switch. This and other problems encountered relating to the performance of TP-336 are discussed in detail in inspection report 87-26.

On May 26, 1987, the design basis reconstitution effort contributed to the identification of a concern involving the interruption of ECCS flow while switching from the injection phase to the recirculation phase following a postulated LOCA. This concern was discussed earlier in paragraph 9.

On June 3, 1987, while troubleshooting the loss of a boron injection path to Unit 4, the ability to inject boron to Unit 3 was also lost. The circumstances surrounding the failure of the boric acid transfer pumps seals and the subsequent nitrogen entrainment of the boric acid transfer pumps and charging pumps is the subject of inspection report 87-28.

On June 8, 1987, the ENS phone was out of service. The cause of the malfunction was not found. AT&T and NRC were notified. Commercial telephone service was verified to be operable.

On June 8, 1987, while Unit 4 was in cold shutdown, the 4C CCW pump automatically started on low header pressure. The cause was attributed to a failed tube in the 4A CCW heat exchanger which subsequently lowered surge tank level and header pressure. The 4A heat exchanger was taken out of service and the failed tube was repaired.

On June 16, 1987, evaluation of CCW heat exchanger performance data identified that on December 11, 1986, ICW inlet temperatures exceeded limits established in the licensees safety evaluation. This subject is addressed in greater detail in paragraph 9.C.