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AUTH. NAME AUTHOR AFFILIATION

PLUNKETT, T. F. Florida Power & Light Co.

RECIP. NAME RECIPIENT AFFILIATION

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SUBJECT: "Annual 10CFR50.59 Rept on Changes Test & Experiments" for 920701-930601. W/931118 ltr.

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L-93-290
10 CFR 50.59 (b)(2)

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket No. 50-250 and 50-251
10 CFR 50.59 Report

Florida Power & Light Company's Report on "Changes, Tests and Experiments Made Without Prior Commission Approval" for the period July 1, 1992 through June 1, 1993 is attached.

Very truly yours,

T. F. Plunkett
Vice President
Turkey Point Nuclear

TFP/GS/rt

Attachment

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
T. P. Johnson, Senior Resident Inspector, USNRC, Turkey Point Plant

IE47

ANNUAL 10 CFR 50.59 REPORT

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 & 4

9312020007

TURKEY POINT PLANT UNITS 3 AND 4

DOCKET NUMBERS 50-250 AND 50-251

CHANGES, TESTS AND EXPERIMENTS

MADE AS ALLOWED BY 10 CFR 50.59

FOR THE PERIOD OF

JULY 1, 1992 THROUGH JUNE 1, 1993

INTRODUCTION

This report is submitted in accordance with 10 CFR 50.59(b), which requires that:

i) changes in the facility as described in the SAR

ii) changes in procedures as described in the SAR, and

iii) tests and experiments not described in the SAR

which are conducted without prior Commission approval be reported to the Commission at least annually. This report is intended to meet this requirement for the period of July 1, 1992, through June 1, 1993.

This report is divided into five (5) sections; the first, changes to the facility as described in the SAR performed by a Plant Change/Modification (PC/M); the second, changes to the facility or procedures as described in the SAR not performed by a PC/M and tests and experiments not described in the SAR; the third, a summary of any fuel reload evaluations; the fourth, a list of Power Operated Relief Valve (PORV) actuations, which is submitted in accordance with FPL's commitment to comply with the requirements of Item II.K.3.3 of NUREG 0737; the fifth, a summary of the findings of Steam Generator tube inspections. Both Unit 3 and Unit 4 had Steam Generator tube inspections during this reporting period.

TABLE OF CONTENTS

SECTION 1

PLANT CHANGE / MODIFICATIONS

PLANT CHANGE/MODIFICATION 84-83

UNIT : 3
TURN OVER DATE : 05/27/93

CATHODIC PROTECTION FOR CCW AND TPCW HEAT EXCHANGERS

Summary:

This design package covered the installation of an impressed cathodic protection system for each of the Turbine Plant Cooling Water (TPCW) Heat Exchangers and the Component Cooling Water (CCW) Heat Exchangers for Turkey Point Unit 3. Included in the modification was the installation of two reference cell electrodes in each of the CCW Heat Exchangers and the TPCW Heat Exchangers. The button anodes originally installed in the TPCW Heat Exchangers channel covers were replaced with probe anodes. No modification was required in regards to the anodes installed in the CCW Heat Exchangers.

Safety Evaluation:

The cathodic protection system was design to improve the longterm useability of the equipment. Its misuse or failure would not hinder the functional ability of the heat exchangers to mitigate the consequences of an accident or to maintain safe shutdown conditions, since sufficient administrative controls existed to control and identify operating problems. In addtion, the impairment of either heat exchanger's functional ability would be a longterm process recognizable during routine maintenance activities. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 85-54

UNIT	:	4
TURN OVER DATE	:	05/28/93

FUEL TRANSFER SYSTEM CABLE DRIVE MODIFICATIONS

Summary:

This design package provided the engineering and design necessary to upgrade the Turkey Point Unit 4 fuel transfer system. The fuel transfer system was modified to provide a more reliable operating system and reduce the amount of equipment under water. This resulted in a more ALARA effective system. These modifications involved the following: (1) fuel transfer system traverse drive source modifications; (2) addition of upender lifting frame counterweights, winch cable and bushings; and, (3) addition of upender winch load monitors and quick-disconnect control consoles.

Safety Evaluation:

As described in Appendix 5A of the Turkey Point Plant Units 3 and 4 Updated FSAR, the fuel transfer system does not perform a safety related function. However, there is a very low probability that a fuel handing accident could result as described in Chapter 14.2 of the Updated FSAR. In order to minimize the effects of these potential events, the fuel handling system modifications were designed to withstand all applicable load combinations, including seismic loads, in accordance with Updated FSAR criteria. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 85-105

UNIT	:	3
TURN OVER DATE	:	02/05/93

MAIN FEEDWATER BYPASS AIR SUPPLY SOLENOID VALVES

Summary:

This modification replaced the existing main feedwater bypass air supply solenoid valves. The existing model of ASCO solenoid valves were replaced with another model of ASCO solenoid valves, which were qualified and had a better temperature rating. The replacement solenoid valves have the same overall dimensions and no system alterations were required.

Safety Evaluation:

The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 85-154

UNIT : 3
TURN OVER DATE : 02/10/93

PROTECTIVE DOORS FOR 3D01 DC DISTRIBUTION PANEL BREAKERS

Summary:

This design package provided for the installation of protective doors on three of the six subpanels of the 3D01 DC distribution panel. This DC distribution panel is located east of the Cable Spreading Room in the Auxiliary Building and east of the Unit 3 Motor Generator sets. These protective doors cover the breakers located in the lower half of the subpanels to prevent the inadvertent closing of these breakers which may cause the unit to trip. The doors were made of expanded metal sheets with a sheet metal frame. The doors were connected to the exterior sheet metal skin of the subpanels using sheet metal tapping screws and three steel hinges. The hinges and fasteners were required to support the deadweight of the door and seismic loads. The stresses induced in the metal door panels and the sheet metal frames were found to be within the allowable capacities of the materials used.

Safety Evaluation:

The installation of protective doors on the 3D01 DC distribution panel did affect the electrical function of the panel and therefore, did not perform a safety related function. However, the doors were designed and installed so as not to inhibit the safety functions of the DC distribution panel itself. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 86-045

UNITS : 3 & 4
TURN OVER DATE : 08/20/92

AUXILIARY FEEDWATER TURBINE EXHAUST SILENCER CONDENSATE REMOVAL

Summary:

The modifications provided in this engineering package will ensure the adequate removal of condensate from the turbine exhaust line, precluding the discharge of hot condensate from the silencers upon AFW pump actuation. The modifications direct the turbine exhaust drain discharge through a 1-1/2" header to an area where it will not be a hazard to personnel. The modifications in this engineering package consisted of locking the existing manual isolation valves in the open position, enlarging the discharge piping downstream of the existing steam orifices, and connecting the discharges to a new 1-1/2 inch drain header. These modifications were required to ensure adequate condensate removal from the turbine exhaust lines to preclude the discharge of hot condensate from the silencers. The existing turbine exhaust drains currently discharge to a drainage trench in the Auxiliary Feedwater pump area. The modifications direct the turbine exhaust piping drain discharge to a storm drain in the Unit 3 Steam Generator Blowdown Tank Area by way of 1-1/2 inch header. The isolation valves were locked in an open position, which will prevent the inadvertent closure of the valve.

Safety Evaluation:

The modifications for connecting existing turbine casing and exhaust drains and the Trip and Throttle valve stem packing high pressure leakoff drains to a new 1-1/2 inch header were required to ensure adequate condensate removal for personnel protection. The modifications did not have an adverse impact on the Auxiliary Feedwater (AFW) System. In the case of the turbine exhaust drains, the AFW turbine was only removed from service within the conditions allowed by the plant Technical Specifications. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 86-79

UNITS : 3 & 4
TURN OVER DATE : 04/24/93

SIMULATOR TRAINING FACILITY

Summary:

This engineering package addressed the addition of the Turkey Point Simulator Training Facility. This facility was constructed in order to satisfy NRC training requirements. The structure provides a facility to train the nuclear plant operators in a simulated control room, as well as, provide training for other operations and maintenance activities. The facility is located on the southwest corner of the site, outside of the plant security fence. The facility is a two story reinforced concrete and masonry structure. The building contains the simulator control room, computer room, classrooms, offices, library and maintenance training areas. Building utilities (power, telecommunications, sanitary, potable water, and fire water) are tied to existing site systems. A paging system designed to extend the Site Evacuation Alarm into the building is also included.

Safety Evaluation:

The Simulator Training Facility engineering package did not modify or affect any plant nuclear safety related systems nor does it perform an automatic nuclear safety related function. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 87-140

UNITS	:	3 & 4
TURN OVER DATE	:	12/16/92

HVAC DUCTWORK TEST HOLE COVERS

Summary:

This engineering modification provided for the installation of HVAC flow test hole covers in the ductwork of all existing HVAC systems in the Units 3 and 4 Containment, Auxiliary Building, Radwaste Building, Fuel Handling Building, Turbine Building and Control Building. The test hole covers provided a 1-1/8 inch access port for a portable flow measurement probe. The addition of test hole covers enhanced the ability to obtain flow measurements, assess HVAC system performance, and allow for proper balancing of the HVAC systems. The installation and location of the test hole covers was performed in accordance with ASHRAE Standards.

Safety Evaluation:

The test hole covers are designed and constructed consistent with existing HVAC systems and ASHRAE standards. A number of factors were considered in the evaluation of this modification. Among them was the increase of bulk material inventory inside the containment which was expected to have a negligible effect on the ECCS heat sink analysis and on the potential for hydrogen generation as stated in Updated FSAR. The modifications in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modifications did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 87-220

UNITS	:	3 & 4
TURN OVER DATE	:	12/21/92

5KV ELECTRICAL PENETRATIONS / MP-0728 LEAK RATE REVISION

Summary:

This engineering package was issued to increase the leak rate acceptance threshold for the 5kV electrical penetrations to make leakage testing practicable. This PC/M evaluated the acceptability of a change in the leakage acceptance criteria and provided the basis for changes in the plant maintenance procedure MP-0728. The leak rate testing performed by MP-0728 is identified in this procedure as a 10 CFR 50, Appendix J "Type B test" and was intended to detect local leaks across the pressure boundary formed by the electrical penetration assembly. It was discovered that the leakage criteria that was originally contained in Maintenance Procedure MP-0728 actually originated from IEEE Std. 317-1983, which prescribes requirements for post-installation testing and was not intended for post-maintenance testing.

Safety Evaluation:

Since the leakage criteria of the IEEE standard is less than one thousands of one percent of the total allowable leakage from all containment penetrations, a nominal increase in the leakage rate of each 5kV electrical penetration was considered negligible compared to the size of the total allowable leak rate specified for all containment penetrations. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 87-245

UNIT : 3
TURN OVER DATE : 05/28/93

VELAN VALVE TAG NO. 3-333 COMPONENT SUBSTITUTION

Summary:

This engineering package provided the engineering basis for a change in material for the packing washer on the Velan valve 3-333. The subject Class 2 safety related valve is in a CVCS charging line to the RCS Loop A Cold Leg that is required to function to provide one means of reactivity control to satisfy NRC requirements. The function of the packing washer, is to precede the packing for effective seating and sealing at the base of the packing gland. Velan originally supplied the subject 3-inch bonnetless bypass valve around HCV-121 with a packing washer made from ASTM-A276 (SS304). Velan recommended a change in the material of this washer to ASTM-A 564 (SS603) and this engineering package was developed to facilitate this change.

Safety Evaluation:

The above part was evaluated as to its appropriate ASTM standard and found to have equal or better strength characteristics as compared to the originally specified material. The differences in chemical composition between the existing and proposed material are negligible, thus, are acceptable. The corrosion resistance of the new material was considered to be comparable to the original material and could be used in the RCS. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 87-279

UNITS	:	3 & 4
TURN OVER DATE	:	07/23/92

ELECTRICAL AND I&C DRAWING SEPARATION AND RENUMBERING

Summary:

The scope of this engineering package was the enhancement of electrical and I&C drawings under a program which maintains traceability to and the continuing fidelity of plant drawings. The drawing enhancement and renumbering provided by this engineering package was initiated with the intent of improving the usefulness of the existing plant drawings for plant personnel. This also was intended to ensure that future modifications were properly designed and implemented, as well as making interpretation of the drawings much easier. To alleviate possible confusion resulting from the continued use of common drawings for the implementation of unit specific design changes, the plant drawings were split into separate drawings for Unit 3 and Unit 4. Additionally, many drawings were further split, such that, only one piece of equipment was depicted on each drawing. The drawings were enhanced by the addition of reference information that was typically missing from the existing plant drawings. Title blocks were standarized so that equivalent information was shown in the same place on every title block. A parallel drawing numbering system was used, such that, a number used on one unit is reserved for the same purpose on the other unit. A drawing cross-reference which ties existing drawing numbers to new drawing numbers was created. Included under the drawing enhancement scope was the incorporation of "as-built" NCRs associated with the Emergency Load Sequencers and Emergency Diesel Generators that were dispositioned during the 1987 Unit 3 and Unit 4 outages. In all instances, the new drawings were prepared and verified by engineering personnel using established project procedures.

Safety Evaluation:

The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 87-409

UNITS	:	3 & 4
TURN OVER DATE	:	05/15/93

INSTALLATION OF SPARE SAFETY INJECTION PUMP MOTOR

Summary:

This engineering package provided for the replacement of any of the four safety injection pump motors at Turkey Point Units 3 and 4 with a spare motor purchased by FPL from Westinghouse. This engineering package ensured that the spare motor could be installed to replace any one of the four existing motors in the event of a failure or for maintenance on an existing motor. The availability of a spare motor will preclude any lengthy outage while an installed motor is required or a new motor is purchased. In addition, analysis was performed and documentation provided to ensure the spare motor meets all seismic and environmental qualification requirements. This engineering package also ensured the interchangeability of the safety injection pump motors among themselves. Motors which have been replaced by the new spare motor may themselves eventually become a spare motor used to replace any of the installed pump motors.

Safety Evaluation:

The safety injection pump motors are Class I equipment that power safety injection pumps, which are intended to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant-accident or main steam line break. The spare motor was purchased to safety related requirements and is seismically and environmentally qualified for its intended application. In addition, the spare motor is capable of meeting the safety injection pump performance characteristics identified in the Updated FSAR. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-087

UNIT	:	4
TURN OVER DATE	:	01/15/93

PRESSURIZER SPRAY VALVE PRESSURE CONTROLLER (PC-444C/D) LOW LIMIT CIRCUIT DEFEAT

Summary:

This engineering package provided for the defeat of the low limit circuit internal to pressurizer spray valve pressure controllers (PC-444C and D). This was accomplished by lifting one lead for this circuit. The pressurizer spray valves were originally equipped with an electro/pneumatic (I/P) converter which converts an input signal (from PC-444C and D) to a corresponding pressure output signal. This pressure signal is then used in conjunction with a valve positioner to precisely control PCV-455A and B. The I/P converter is located inside containment. Due to ambient temperature variations, the I/P converter had a tendency to drift causing air to be supplied to the spray valves continuously. This air supply caused the spray valves to remain open even with a zero percent open signal present.

Safety Evaluation:

The pressurizer spray valve controls are not required for any design basis event, do not perform a safety related function, and do not interface with safety related systems or components. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-099

UNIT	:	3
TURN OVER DATE	:	03/16/93

PRIMARY WATER STORAGE DEAERATED WATER TRANSFER PUMP HI LEVEL START SWITCH MODIFICATION

Summary:

This engineering modification retagged and repulled cables for level switches LS-1529 and LS-1532, located on the primary water storage deaerator. This was implemented to correct a mismatch in relative elevations of the deaerator level switch (associated with LT-1532) versus the deaerator water transfer pump start-signal level switch (LS-1529). The elevation of level switch LS-1529 was higher than the upper range of the deaerator level transmitter (LT-1532), which prevented the transfer pump from starting automatically without installing a temporary jumper cable. The elevation of level switch LS-1529 was also near the vacuum pump high level trip (LS-1552). After this modification, the proper automatic operating scheme for the deaerator was restored, with the water level in the primary water storage deaerator rising above the elevation require to start the transfer pumps without causing the vacuum pump to trip.

Safety Evaluation:

The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-100

UNIT : 4
TURN OVER DATE : 03/16/93

PRIMARY WATER STORAGE DEAERATED WATER TRANSFER PUMP HI LEVEL START SWITCH MODIFICATION

Summary:

This engineering modification retagged and repulled cables for level switches LS-1529 and LS-1532, located on the primary water storage deaerator. This was implemented to correct a mismatch in relative elevations of the deaerator level switch (associated with LT-1532) versus the deaerator water transfer pump start-signal level switch (LS-1529). The elevation of level switch LS-1529 was higher than the upper range of the deaerator level transmitter (LT-1532), which prevented the transfer pump from starting automatically without installing a temporary jumper cable. The elevation of level switch LS-1529 was also near the vacuum pump high level trip (LS-1552). After this modification, the proper automatic operating scheme for the deaerator was restored, with the water level in the primary water storage deaerator rising above the elevation require to start the transfer pumps without causing the vacuum pump to trip.

Safety Evaluation:

The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-346

UNIT : 4
TURN OVER DATE : 05/10/93

TURBINE PLANT COOLING WATER ISOLATION VALVE MODIFICATION

Summary:

This modification provided for the addition of controls to Turbine Plant Cooling Water (TPWC) Isolation Valves 50-4-314 and 50-4-334, such that, these valves would close on a Safety Injection Actuation Signal (SIAS). TPCW isolation valves with pneumatic operators were installed to replace manually operated valves by an earlier engineering modification. However, the early modification did not provide for the connection of the valve control circuits. This engineering package installed the controls; instrument air and electrical control circuits to operate existing pilot solenoid valves. Automatic isolation of TPCW during accident conditions ensures required Intake Cooling Water (ICW) flow to the Component Cooling Water (CCW) heat exchangers. Consequently, the modification resolved the single failure concerns associated with valve CV-4-2201 as described in Justification for Continued Operation (JCO) 86-003 and provided a basis for eliminating Unit 4 from the corrective action requirements of this JCO.

Safety Evaluation:

Isolation of the TPCW system occurs following a design basis accident (DBA) to ensure adequate Intake Cooling Water (ICW) flow is diverted to the Component Cooling Water (CCW) heat exchangers for design basis accident heat load removal in the event of a single failure that results in one ICW pump being available. Valves 50-4-314 and 50-4-334 are spring-to-close, fail-closed on loss of air and electrical power, and are provided with 125 VDC pilot solenoid valves. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-450

UNIT : 4
TURN OVER DATE : 04/12/93

RCP MOTOR REFURBISHMENT AND UPGRADE

Summary:

This engineering package provided for the refurbishment and upgrade of the 4B reactor coolant pump motor. The design bases established in the Updated FSAR were reviewed and determined to be unaffected, because thee modifications met all FSAR criteria stipulated for the original design. In addition, these modifications did not impact any Technical Specifications. The original installed motor was replaced with a spare motor which was refurbished at the Westinghouse Electro-Mechanical Division facility. This refurbishment consisted of inspection and maintenance activities performed to the existing design specifications. In addition, two upgrade modifications were performed, concurrent with the refurbishment, to ensure consistency with the latest RCP technology and to realize additional reliability and availability. These modifications consisted of an upgrade to the oil lift system and a redesign of the lower cooling coil. In the past, the lower cooling coil had been susceptible to handling damage due to the use of bronze flanges, copper pipes, and soldered/brazed joints. These were replaced with steel pipe and fittings with welded joints and heavier 90/10 copper/nickel cooling coil tubing. The oil lift system upgrade included such improvements as stainless steel lines, flow control valves, elimination of the 3-way valve system pressure switch settings change, and an enhanced oil lift pump.

Safety Evaluation:

This package is classified as safety related, since it performs work on the lower bearing cooling coil which is considered part of the safety related CCW system. The design bases established in the Updated FSAR were reviewed and determined not to be affected, because these modifications meet all Updated FSAR criteria stipulated for the original design. In addition, these modifications did not impact any Technical Specifications. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-527

UNITS : 3 & 4
TURN OVER DATE : 07/23/92

RESOLUTION OF DRAWING CHANGES ASSOCIATED WITH 5610-T-E-4503

Summary:

This engineering package provided a basis to evaluate and resolve the existing outstanding Requests for Engineering Assistance (REAs) and Non-Conformance Reports (NCRs) associated with Units 3 and 4 safety related systems. In this way, discrepancies between the as-built condition of the plant and the existing Plant Operating Documents were reconciled. This engineering package documented drawing changes associated with 5610-T-E-4503, Sheet 1, which resulted from discrepancies identified by three REAs and an NCR. The changes requested by the REAs included in this package only involved redesignation of valve type or classification.

Safety Evaluation:

The engineering evaluation for this modification concluded that the subject would not alter the plant's design basis and were bounded by existing design analysis. Further, there would be no adverse effect on the plant's systems, structures or components. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-534

UNITS	:	3 & 4
TURN OVER DATE	:	12/29/92

DRAWING DISCREPANCIES ON 5610-T-E-4534, SHEETS 1 AND 2 CONTAINMENT VENTILATION SYSTEM

Summary:

This engineering package was developed to correct drawing discrepancies identified on Operating Diagram 5610-T-E-4534, Sheets 1 and 2. A field verification walkdown of the Containment Ventilation System and a review of Operating Diagram 5610-T-E-4534, Sheets 1 and 2, revealed several drawing discrepancies. These discrepancies involved valve positions for normal operation, the lack of a drawing showing service air, HVAC damper and ductwork locations, flow direction discrepancies, and valve-type discrepancies. The changes to resolve all discrepancies were evaluated by Engineering using nonconformance reports and found to be acceptable.

Safety Evaluation:

Based on an evaluation contained in this engineering package, these discrepancies were determined not to adversely impact plant systems, structures or components. Further, these drawing changes did not alter the plant's design basis and were bounded by existing design analyses. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-095

UNIT	:	4
TURN OVER DATE	:	08/18/92

DRAWING UPDATE 5610-T-E-4065, SHEETS 2 AND 3
LUBE WATER AND CIRCULATING WATER SYSTEM

Summary:

This engineering package was developed to correct drawing discrepancies on Operating Diagram 5610-T-E-4065, Sheets 2 and 3. A field verification walkdown of the Unit 4 Lube Water and Circulating Water Systems and a review of Operating Diagram 5610-T-E-4065, Sheets 2 and 3, revealed several drawing discrepancies. These discrepancies involved setpoints, value positions, test connection locations, and valve types. The changes to resolve all discrepancies were evaluated by Engineering using nonconformance reports and found to be acceptable.

Safety Evaluation:

Based on an evaluation contained in this engineering package, these discrepancies were determined not to adversely impact plant systems, structures or components. Further, these drawing changes did not alter the plant's design basis and were bounded by existing design analyses. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-100

UNIT : 4
TURN OVER DATE : 07/23/92

DRAWING UPDATE 5610-T-E-4061, SHEET 1
LUBE WATER AND CIRCULATING WATER SYSTEM

Summary:

This engineering package was developed to correct drawing discrepancies on Operating Diagram 5610-T-E-4061, Sheet 1. A field verification walkdown of the Unit 4 Main Steam System and a review of Operating Diagram 5610-T-E-4061, Sheet 1, revealed several drawing discrepancies. These discrepancies involved steam trap identification tagging, steam trap isolation and drain valves, small valves tagging designations, and small bore piping and valve configuration for the Moisture Separator Reheater nitrogen system. The changes to resolve all discrepancies were evaluated by Engineering using nonconformance reports and found to be acceptable.

Safety Evaluation:

Based on an evaluation contained in this engineering package, these discrepancies were determined not to adversely impact plant systems, structures or components. Further, these drawing changes did not alter the plant's design basis and were bounded by existing design analyses. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-475

UNITS	:	3 & 4
TURN OVER DATE	:	04/23/93

DRAWING DISCREPANCIES ON 5610-T-E-4501, SHEET 1
REACTOR COOLANT

Summary:

This engineering package was developed to correct drawing discrepancies identified on Operating Diagram 5610-T-E-4501, Sheet 1. A field verification walkdown of small bore piping for the Reactor Coolant System and a review of Operating Diagram 5610-T-E-4501, Sheet 1, revealed several drawing discrepancies. These discrepancies involved incorrectly labeled valves; the exact locations of several small valves, blind flanges, and instrument taps; and piping caps. The changes to resolve all discrepancies were evaluated by Engineering using nonconformance reports and were found to be acceptable.

Safety Evaluation:

Based on an evaluation contained in this engineering package, these discrepancies were determined not to adversely impact plant systems, structures or components. Further, these drawing changes did not alter the plant's design basis and were bounded by existing design analyses. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-491

UNITS	:	3 & 4
TURN OVER DATE	:	01/11/93

STRUCTURAL STEEL ATTACHMENT FIREPROOFING REQUIREMENTS

Summary:

This engineering package reevaluated the fireproofing requirements applicable to structural steel attachments that penetrate the fireproofing envelope and implemented drawing changes which required modifications to the structural steel fireproofing in four rooms. This modification also provided a basis for changes in plant inspection and maintenance procedures associated with structural steel fireproofing requirements. During the periodic reinspections of structural steel fireproofing, as required by Plant Procedure O-SMM-016.3 "Fire Barriers and Structural Steel Fireproofing Inspection," questions were raised relative to implementation requirements for fireproofing attachments which penetrate the structural steel fireproofing envelope. Although details for attachments are shown on drawings 5610-A-181, Sheet 1 and 2, the requirements for attachments penetrating fireproofing were reevaluated and clarified to ensure a consistent treatment of all cases. A compartment heat load analysis was performed for all rooms containing fireproofed structural steel.

Safety Evaluation:

The fireproofing material specified was identical to that originally installed. The changes which were made in this engineering package were determined not to adversely impact plant systems, structures, or components. Furthermore, these changes did not alter the plant's overall design basis, which was bounded by existing design analyses. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-512

UNITS : 3 & 4
TURN OVER DATE : 05/13/93

NUCLEAR OPERATIONS/CHEMISTRY BREAK AREA AND CONTROL POINT

Summary:

This engineering package provided a break area/control point for Nuclear Operations (NO) and Chemistry personnel, since they are not permitted to eat inside the Radiation Control Area (RCA). Locating a shelter outside the RCA, in the Turbine Building/yard area on the west side of the existing RCA fence, provided a convenient RCA entry/exit point for NO and Chemistry personnel as well as a break area. This shelter was constructed as a pre-fabricated non-combustible shelter. Additional fencing was provided to direct personnel from the access point to the break area/control point. This required the installation of a personnel contamination monitor and a hand frisker. Both pieces of equipment were relocated from the existing control point/guard shack located north of this new control point. Access to and from the Radiation Control Area (RCA) through the original control point/guard shack was no longer permitted, and access was transferred to this new control point.

Safety Evaluation:

This shelter and related components do not provide any nuclear safety functions. The shelter was located in a relatively clear area of the Turbine Building/yard area, and was designed to withstand the wind and roof loading requirements of the South Florida Building Code. The shelter and related components were reviewed for the seismic requirements. These design basis requirements were implemented to preclude any potential interaction with future safety related systems, structures, and components. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-542

UNITS	:	3 & 4
TURN OVER DATE	:	05/10/93

DRAWING UPDATE - ISI QUALITY GROUP CLASSIFICATIONS/BOUNDARIES

Summary:

This engineering package provided a basis for correcting design document deficiencies within the existing series of quality group classification ("Code Boundary") drawings that were identified during previous QA audits. This engineering package corrected these deficiencies by adding the quality group classifications/boundaries on selected POD T-E documents which best represent present day plant configurations and by providing bases for classifications/boundaries based on current regulatory commitments and other guidelines. This plant change did not involve any physical plant configuration change. The accurate, up to date drawings were required in order to properly establish pressure test program requirements and appropriate procedures for In-service Inspection and Testing. This included ASME Section XI ISI code testing and support design requirements for repair and modifications to the piping systems.

Safety Evaluation:

The Quality Group Classification bases evaluated in this engineering package were not considered an operability concern based on design equivalence. They did not impact system operation or create any safety related concerns. They were drawing additions of Quality Group Classifications which were evaluated to be acceptable. No new equipment or components were installed. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

UNIT : 3
TURN OVER DATE : 10/08/92

ADDITION OF APPENDIX R BYPASS SWITCH FOR LCV-3-460

Summary:

This engineering package provided a keylocked bypass switch located on the main control board 3C03, which defeated the electrical interlock between Chemical Volume Control System (CVCS) valves LCV-3-460 and CV-3-200A, B, and C. The bypass switch would be used only in the event that a fire causes a hot short to spuriously open one of the CV-3-200A, B, or C valves and prevent closure of LCV-3-460. The addition of the bypass switch replaced the original requirement for pulling the control fuses associated with CV-3-200A, B, and C to defeat the circuit interlock. These modifications ensured the availability of LCV-3-460 to perform its safe shutdown function for postulated fire scenarios causing spurious opening of CV-3-200A, B, and C.

Credit is taken during certain Appendix R fire scenarios, including Alternate Shutdown, for LCV-3-460 to provide CVCS letdown isolation during safe shutdown. LCV-3-460 is a DC solenoid controlled valve which has circuit interlocks with downstream orifice isolation valves, CV-3-200A, B, and C. This interlock is intended to prevent potential damage to the regenerative heat exchanger and relief valve RV-3-203 due to pressure transients in the line between LCV-3-460 and the CV-3-200 valves. CV-3-200A, B, and C are DC solenoid-controlled valves that close on loss of electrical power or loss of control air. Spurious opening of any one of the CV-3-200 valves due to a hot short would prevent closure of LCV-3-460, because of the electrical interlock between the valves. This condition was not a concern for Alternate Shutdown but was valid for other fire zones. This engineering package modification served to correct this potential issue.

Safety Evaluation:

This modification enhanced the capability of the control room operator to defeat the interlock between LCV-3-460 and CV-3-200A, B, and C and mitigate the consequences of a fire in areas outside the alternate shutdown areas caused by a hot short. The operators

ADDITION OF APPENDIX R BYPASS SWITCH FOR LCV-3-460

ability was enhanced by providing a bypass switch, located adjacent to the existing control switch for LCV-3-460 to defeat the interlock. This bypass switch provided the operator a quicker and more desirable method to mitigate the consequences of a fire and is also consistent with NRC guidance for actions required to achieve hot shutdown. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-194

UNIT : 4
TURN OVER DATE : 04/19/93

ADDITION OF APPENDIX R BYPASS SWITCH FOR LCV-4-460

Summary:

This engineering package provided a keylocked bypass switch located on the main control board 4C03, which defeated the electrical interlock between Chemical Volume Control System (CVCS) valves LCV-4-460 and CV-4-200A, B, and C. The bypass switch would be used only in the event that a fire causes a hot short to spuriously open one of the CV-4-200A, B, or C valves and prevent LCV-4-460 closure. The addition of the bypass switch replaced the original requirement for pulling the control fuses associated with CV-4-200A, B, and C to defeat the circuit interlock. These modifications ensured the availability of LCV-4-460 to perform its safe shutdown function for postulated fire scenarios causing spurious opening of CV-4-200A, B, and C.

Credit is taken during certain Appendix R fire scenarios, including Alternate Shutdown, for LCV-4-460 to provide CVCS letdown isolation during safe shutdown. LCV-4-460 is a DC solenoid controlled valve which has circuit interlocks with downstream orifice isolation valves, CV-4-200A, B, and C. This interlock is intended to prevent potential damage to the regenerative heat exchanger and relief valve RV-4-203 due to pressure transients in the line between LCV-4-460 and the CV-4-200 valves. CV-4-200A, B, and C are DC solenoid-controlled valves that close on loss of electrical power or loss of control air. Spurious opening of any one of the CV-4-200 valves due to a hot short would prevent closure of LCV-4-460, because of the electrical interlock between the valves. This condition was not a concern for Alternate Shutdown but was valid for other fire zones. This engineering package modification served to correct this potential issue.

Safety Evaluation:

This modification enhanced the capability of the control room operator to defeat the interlock between LCV-4-460 and CV-4-200A, B, and C and mitigate the consequences of a fire in areas outside the alternate shutdown areas caused by a hot short. The operators

ADDITION OF APPENDIX R BYPASS SWITCH FOR LCV-4-460

ability was enhanced by providing a bypass switch, located adjacent to the existing control switch for LCV-4-460 to defeat the interlock. This bypass switch provided the operator a quicker and more desirable method to mitigate the consequences of a fire and is also consistent with NRC guidance for actions required to achieve hot shutdown. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-239

UNIT : 4
TURN OVER DATE : 05/10/93

C BUS SWITCHGEAR CONTROL AND PROTECTION POWER ISOLATION FOR APPENDIX R

Summary:

This modification provided for the installation of a molded case circuit breaker at the C Bus switchgear to connect or disconnect the control and protection power for the switchgear. This eliminated the present fuse pulling requirement and, thereby, reduced the operator burden associated with supplying power to the Standby Steam Generator Feedwater (SSGF) pumps. The C Bus switchgear provides power to the Standby Steam Generator Feedwater Pumps (SSGFPs). These pumps provide an alternate source of feedwater to the steam generators. In the event of a postulated fire which could render the Auxiliary Feedwatr System (AFW) inoperable the SSGF pumps are utilized to provide feedwater. During this condition credit is taken to power the C Bus switchgear from the Units 1 and 2 Cranking Diesels after tripping the C Bus switchgear breakers. The required safe shutdown breakers are then manually aligned.

Safety Evaluation:

The modification provided by this engineering package installed a molded case circuit breaker on the switchgear door. The use of a breaker reduced the number of actions required to connect or disconnect the C Bus switchgear control and protection power. In addition, since the breaker was flush mounted to the switchgear door, operator entry into the panel to pull fuses was eliminated. Therefore, this modification enhanced the means by which the control and protection power was isolated. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-240

UNIT : 3
TURN OVER DATE : 11/17/92

RAD-3-6417 SAMPLE LINE END CABINET MODIFICATION

Summary:

This engineering package eliminated the chillers and drain tanks located in the Unit 3 Steam Jet Air Ejector exhaust system. The chiller system had been installed to remove entrained water and water vapor from the exhaust flow prior entering radiation monitor RaD-3-6417. The original equipment was incompatible with the exhaust constituents (a caustic mixture of air, water, and ammonial), which had resulted in a history of material deterioration, failure and excessive maintenance requirements. The original chillers, drums, connecting piping and associated equipment were removed. The sample line to RaD-3-6417 was re-routed to take the sample from the existing (plugged) threaded connection on the air ejector exhaust gooseneck. The sample line from the existing water separator to the monitor was heat traced and insulated to heat the sample prior to its passage through the monitor. This decreased the sample relative humidity and prevented condensation in the detectors, without the complications and maintenance requirements imposed by the existing chiller system.

Safety Evaluation:

The modification increased the reliability of the subject monitor, which decreased the likelihood of entering a Technical Specification Action statement. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-396

UNIT	:	3
TURN OVER DATE	:	10/16/92

NIS RECORDER CHANNEL SELECTOR SWITCHES

Summary:

This engineering package involved the modification of the NIS recorder channel selector switches in the control room. This modification consisted of replacing the existing twelve position NIS recorder channel selector switches with eight position switches. In addition, wiring associated with these four unused switch positions located between the first terminal block in panel 3C01 and the selector switches was removed. During the detailed control room design review, the unused positions of the nuclear instrumentation system recorder selector switches were identified as a human engineering deficiency (HED No. TA-40). Florida Power and Light committed to resolve this HED by eliminating the unused switch positions and changing the escutcheon plates.

Safety Evaluation:

The replacement eight position switches were equivalent in all respects to the existing twelve position switches. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-397

UNIT : 4
TURN OVER DATE : 05/03/93

NIS RECORDER CHANNEL SELECTOR SWITCHES

Summary:

This engineering package involved the modification of the NIS recorder channel selector switches in the control room. This modification consisted of replacing the existing twelve position NIS recorder channel selector switches with eight position switches. In addition, wiring associated with these four unused switch positions located between the first terminal block in panel 4C01 and the selector switches was removed. During the detailed control room design review, the unused positions of the nuclear instrumentation system recorder selector switches were identified as a human engineering deficiency (HED No. TA-40). The nuclear instrumentation system recorder provides the control room operator with trending information. This data is particularly valuable during reactor startup and other power transients. If these selector switches were inadvertently placed in one of the unused positions the resulting display could confuse the control room operator. Florida Power and Light committed to resolve this HED by eliminating the unused switch positions and changing the escutcheon plates.

Safety Evaluation:

The replacement eight position switches were equivalent in all respects to the existing twelve position switches. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-445

UNIT : 3
TURN OVER DATE : 11/16/93

DRILLING OF VALVE WEDGE FOR MOV-3-872

Summary:

This engineering modification provided for the drilling of a small pressure relieving hole in the valve wedge on the Reactor Coolant System (RCS) side of MOV-3-872. MOV-3-872 is a component of the Alternate Low Head Safety Injection flowpath and is classified as safety related. As documented in INPO SOER 84-7, system pressure in the valve bonnet area may become trapped causing a high differential pressure across the valve disc/wedge and resultant binding during valve opening. These INPO reported failures have prevented safety related systems from functioning when called upon to operate. A subsequent engineering analysis determined that MOV-3-872 is not affected, however, as a long-term precautionary measure it was recommended that MOV-3-872 be modified to prevent the potential for pressure locking. This modification eliminated the potential for such binding.

Safety Evaluation:

This modification did not affect the function of MOV-3-872 nor the operation of any plant systems. INPO SOER 84-7 documented the disc drilling modification as an acceptable solution to the potential for pressure locking and Velan concurred with drilling location and size of the hole. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-446

UNITS	:	3 & 4
TURN OVER DATE	:	09/23/92

WATER TREATMENT PLANT IN-LINE MONITORS

Summary:

This engineering package provided for the modifications of the Nuclear Chemistry Building and Water Treatment Plant to address INPO Finding CY.3-2. The modifications consisted of routing tubing from the cation, anion and mixed bed demineralizers and final effluent station to the Nuclear Chemistry Building, and connecting service water and drains to the Nuclear Chemistry Building lab sink. These modifications provided a central location for water chemistry analysis of the demineralized water produced by the Water Treatment Plant.

Safety Evaluation:

The modifications performed by this engineering package involved the Water Treatment Plant System. There were no safety related systems affected by the implementation of this engineering package. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-449

UNITS	:	3 & 4
TURN OVER DATE	:	08/13/92

CCW AREA PIPE TRENCH FLOODWALLS

Summary:

This engineering package provided engineering and design required to install new reinforced concrete floodwalls, and repair existing floodwalls in pipe trenches located on the east side of the Auxiliary Building in the Component Cooling and Safety Injection areas. Technical Issue No. 4 under the FPL Systematic Design Investigation (SDI) Program identified four pipe trenches, located on the east side of the Auxiliary Building in the Component Cooling and Safety Injection areas, as potential points of flood water intrusion into the flood protected area in the event of a hurricane surge tide. Flood water intrusion into the plant could adversely affect equipment or components important to safety. The new floodwalls were installed in the Unit 3 and 4 component cooling pipe trenches, and the existing floodwalls in the Unit 3 and 4 safety injection pipe trenches had gaps sealed. Flexible pressure boots were installed at large bore pipe penetrations in the component cooling water pipe trenches to provide a barrier against flood water intrusion while allowing for the design pipe movement.

Safety Evaluation:

The effects of this modification on the external flood protection system were reviewed and no adverse effects will result from this implemented modification. The ability of the external flood protection system to perform its design function was enhanced by the installation and repair of floodwalls. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 91-128

UNIT : 3
TURN OVER DATE : 10/20/92

480V UNDERVOLTAGE PROTECTION SCHEME MODIFICATION

Summary:

This engineering package modified the 480V load center non-safety injection degraded voltage schemes. These modifications were required to improve the repeatability of the existing degraded voltage relays to their specified setpoints. The original logic did not provide detection of auxiliary relay coil failures, and did not allow the circuit to be placed in the trip mode under the conditions of such an event. Redesign of the original degraded voltage scheme mitigated these conditions. This modification also installed a bypass switch which will allow one channel of the degraded voltae scheme to be placed in the trip mode when one or both of the relays of that channel are removed for relay testing or calibration. To accommodate the modifications provided by this engineering package, provisions for a minor change to the Technical Specifications were instituted. This change deleted the specific reference to the inverse time relay. A HOLD POINT was placed on this engineering package to restrict any changes to the plant configuration as described in the Technical Specification until NRC approval was received.

Safety Evaluation:

This modification was evaluated under the requirements of 10 CFR 50.59 and did not constitute an unreviewed safety question. This activity does not change the operation, function or design bases of any structure, system or component important to safety as described in the SAR. In particular, the undervoltage protection scheme still used relays to detect a degraded voltage condition and to actuate sequencer trip using a two-out-of-two logic. The Technical Specification requirements applicable to this modification were not affected. However, the method of satisfying the Technical Specification requirements was affected. While the setpoint values are unchanged, the means of satisfying these setpoint requirements was accomplished by a new definite time delay undervoltage relay. The analysis performed to support this circuit modification confirmed that the new relays, at the existing setpoints, provided the necessary degraded voltage protection.

PLANT CHANGE/MODIFICATION 91-130

UNIT : 3
TURN OVER DATE : 10/21/92

PROCESS RADIATION MONITORING SYSTEM R-3-11 AND R-3-12 REPLACEMENT

Summary:

This engineering package was issued to replace containment particulate and noble gas radiation monitors R-3-11 (particulate) and R-3-12 (gaseous) and associated displays and controls in control room panel 3QR66 with new ones, which were expected to be more reliable and easier to maintain. R-3-11 and R-3-12 provide a means for monitoring the Unit 3 containment atmosphere for radioactivity released from normal operation, anticipated transients, and accident conditions. These instruments are part of the Engineered Safety Features Instrumentation. The new equipment performs similar functions to the original equipment with the exception of the capability to monitor the common plant vent. Since R-14 and RaD-6304 monitored the common plant vent, the need for R-3-11 and R-3-12 to monitor the common plant vent was no longer necessary and was deleted.

Safety Evaluation:

After the monitor replacements, monitors R-3-11 and R-3-12 continue to perform their safety functions, whereby the high radiation level for the channel initiates closure of the containment purge supply and exhaust duct valves and instrument air bleed valves, and initiates control room ventilation isolation, as described in the Updated FSAR. The provisions for monitoring the plant vent using R-3-11 and R-3-12 was no longer necessary, since this function continued to be accomplished by monitors R-14 and RaD-6304. Wide range monitor RaD-6304 was used to satisfy the monitoring requirements of Regulatory Guide 1.97. In addition, monitors R-3-11 and R-3-12 provided early detection capability, since containment atmosphere is monitored directly. These functions fulfilled the leakage detection system requirements and the Engineered Safety Features Instrumentation requirements as described in the Updated FSAR. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 91-133

UNIT : 3
TURN OVER DATE : 07/31/92

REPLACEMENT OF 480 VOLT MOTOR CONTROL CENTER 3E

Summary:

This Engineering Package provided the engineering and design necessary to replace MCC 3E with a new Motor Control Center. Rust and corrosion of the internal structure of the non-nuclear safety related 480 Volt Motor Control Center (MCC) 3E, located outdoors at the intake structure, had resulted in structural degradation extensive enough to require replacement. The cause of this degradation was related to the utilization of noncorrosion-resistant materials in the original design. Also, the MCC was obsolete and obtaining replacement parts was becoming difficult. The new MCC was designed and constructed as a standard MCC installed in a stainless steel (type 304L) NEMA 4X enclosure, providing increased corrosion protection. Moisture entry from a manhole below was blocked by a stainless steel bottom on the enclosure and cable entry openings were sealed after installation of the cables.

Safety Evaluation:

Motor Control Center 3E does not supply power to or control any nuclear safety related plant equipment and is normally powered, via MCC 3F, from Load Center 3F, which in turn is powered from non-safety related 4160 VAC Switchgear 3C. None of the equipment in the vicinity of MCC 3E was Nuclear Safety Related. MCC 3E cannot be powered from the safety related emergency diesel generators, and is not powered from the vital 125 VDC system. The modifications in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 91-166

UNIT : 3
TURN OVER DATE : 11/29/92

REPLACEMENT OF SEAL TABLE FITTINGS AND THIMBLE TUBE LENGTHENING

Summary:

This engineering modification provided for the replacement of the existing seal table fittings with new fittings that contained an integral low pressure seal for refueling. The original fittings were frequently a source of primary leakage during plant startup and operation. The original low pressure refueling seals were difficult to assemble and often leaked.

The new design eliminated the original guide tube ferrule which was frequently the location of leakage when the seal was configured for plant operation. The replacement design was welded to the guide tube and had a tapered machined sealing surface that replaced the ferrule. The new refueling seal utilized the same sealing technique as the original seal, i.e., compression of an elastomer to form the seal. The difference in the refueling seal was that the new fitting had the seal permanently installed and used an internal nut for compression. This modification also provided for lengthening of thimble tubes to compensate for reduced core inserted thimble end elevations. This condition was associated with new fitting stackup dimensions and previous thimble tube shortening activities.

Safety Evaluation:

The new seal table fittings, thimble tube extensions and replacement guide tubes were similar to the existing hardware and incorporate some improvements. The capping abandoned thimble tubes and provided an improved and more reliable seal design than the original isolation valve, in the event of a thimble tube leak inside the RCS pressure boundary. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 91-198

UNIT	:	3
TURN OVER DATE	:	11/14/92

REPAIR AND MODIFICATION OF THE UNIT 3 INTAKE STRUCTURE

Summary:

This engineering package restored, as required, the concrete slabs supporting Unit 3 ICW 3A and 3B Pumps and the Screen Wash Pumps to a condition which met the original design bases, and ensured acceptable long-term performance. This was accomplished for the ICW 3A and Screen Wash Bays by removing deteriorated concrete and reinforcing steel, protecting uncovered reinforcing steel from future corrosion, and replacing concrete with material of equal or greater strength. Repairs were implemented using Nonconformance Reports, which were then evaluated and dispositioned by Engineering providing appropriate repairs for each bay.

Safety Evaluation:

Upon completion of the modifications, the structural integrity of the Intake Structure slabs were restored to withstand all applicable loads in accordance with the requirements for Class I structures identified in the Updated FSAR, including operating loads of the pumps and associated components, thereby meeting the original design intent of the slab. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-004

UNITS	:	3 & 4
TURN OVER DATE	:	05/25/93

UPGRADING PLANT PAGE AUDIBILITY

Summary:

This engineering package provided the design to supplement the plant page alarms in the high noise areas by replacing 13 existing blue lights with high intensity strobe lights and also added 31 high intensity strobe lights at various locations. The Turkey Point Public Address System provides normal plant paging capabilities and is also utilized to broadcast site evacuation and containment evacuation alarms throughout the plant. The new strobe lights would be activated during any emergency alarm. The new strobe lights were powered from 120 VAC paging power supply. This improved awareness to the emergency alarm in all plant areas. The original separate fire alarm system wiring was abandoned and fire horns and fire alarm control relays were removed. The new tone generator would broadcast the fire alarm over the existing public address system speakers.

Safety Evaluation:

No credit is taken for the public address system to support operator actions to accomplish safe shutdown or accident mitigation, to prevent uncontrolled release of radioactivity or to perform a fire protection function. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-033

UNIT : 3
TURN OVER DATE : 10/21/92

EMERGENCY BUS LOAD SEQUENCER MODIFICATIONS

Summary:

This engineering package modified the Unit 3 programmable emergency bus load sequencers to eliminate the root cause of the failure that resulted in Sequencer 4A aborting an Auto Test and to address the results of the Failure Mode and Effects Analysis identified in Engineering Report No. JPN-PTN-SEIS-92-010. This engineering package also upgraded the sequencers by installing a new output module for diagnostic purposes. The intent of the diagnostics was to create an error message code and to provide additional information to facilitate troubleshooting. Other modifications included in the scope of this engineering package consisted of programming modifications to eliminate a nuisance alarm and the delay (eleven cycles) of the signal from the Auxiliary Transformer breaker position and rewiring of some of the blocking relays to ensure that failure of a relay to de-energize would be displayed in the front panel.

Safety Evaluation:

The sequencer modifications were tested in both the simulator and in the plant sequencers and demonstrated that the sequencer safety functions had not been affected and confirmed the proper interactions between the modified sequencers and plant equipment. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-034

UNIT : 4
TURN OVER DATE : 05/03/92

EMERGENCY BUS LOAD SEQUENCER MODIFICATIONS

Summary:

This engineering package modified the Unit 4 programmable emergency bus load sequencers to eliminate the root cause of the failure that resulted in Sequencer 4A aborting an Auto Test and to address the results of the Failure Mode and Effects Analysis identified in Engineering Report No. JPN-PTN-SEIS-92-010. This engineering package also upgraded the sequencers by installing a new output module for diagnostic purposes. The intent of the diagnostics was to create an error message code and to provide additional information to facilitate troubleshooting. Other modifications included in the scope of this engineering package consisted of programming modifications to eliminate a nuisance alarm and the delay (eleven cycles) of the signal from the Auxiliary Transformer breaker position and rewiring of some of the blocking relays to ensure that failure of a relay to de-energize would be displayed in the front panel.

Safety Evaluation:

The sequencer modifications were tested in both the simulator and in the plant sequencers and demonstrated that the sequencer safety functions had not been affected and confirmed the proper interactions between the modified sequencers and plant equipment. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-040

UNIT : 3
TURN OVER DATE : 12/01/92

ADDITION OF REVERSE POWER RELAY AND MAIN GENERATOR PROTECTION MODIFICATIONS

Summary:

This engineering package modified and upgraded the Turkey Point Unit 3 main generator protection schemes. These modifications realigned the existing main generator protection schemes to mitigate the effects of a partial loss of vital DC power on their protection capability. These modifications included the realignment of the vital DC control power supplying these schemes and addition of a reverse power relay. In addition, this modification upgraded the existing generator protection by regrouping existing main generator primary and backup protective functions, adding an out-of-step relay and main generator output circuit breaker emergency trip control switch. These design upgrades provided additional backup capability and enhancements to the generator protection schemes. The INPO Significant Operating Experience Report (SOER) 81-15 made specific recommendations regarding the ability of plants to manage and recover from a loss of a vital DC bus. Implementation of these modifications satisfied INOP SOER 81-15 recommendation 1C.

Safety Evaluation:

These modifications to the main generator protection schemes enhanced and upgraded existing non-safety related main generator protection system, which was not required for safe shutdown during a design basis accident. The generator protection system remained functionally the same as a result of this modification, and the effects of a failure of any relay remained unchanged from the original design. The safety related function of the 125V vital DC Buses 3D01 and 4323 which feed these circuits were unaffected by this modification, since any power feed realignment performed would be downstream from the 125V vital DC bus isolation circuit breakers. The function of the auxiliary relays, which interface with the safety related diesel generator, remained unchanged and unaffected by this modification. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-054

UNIT	:	4
TURN OVER DATE	:	05/01/93

480V UNDERVOLTAGE PROTECTION SCHEME MODIFICATION

Summary:

This engineering package modified the 480V load center non-safety injection degraded voltage schemes. These modifications were required to improve the repeatability of the existing degraded voltage relays to their specified setpoints. The original logic did not provide detection of auxiliary relay coil failures, and did not allow the circuit to be placed in the trip mode under the conditions of such an event. Redesign of the original degraded voltage scheme mitigated these conditions. This modification also installed a bypass switch which would allow one channel of the degraded voltage scheme to be placed in the trip mode when one or both of the relays of that channel are removed for relay testing or calibration. To accommodate the modifications provided by this engineering package provisions for a minor change to the Technical Specification were instituted. The Technical Specification change deleted the specific reference to the inverse time relay. This change was approved by the NRC as Amendments No. 152 and No. 147 for Units 3 and 4, respectively.

Safety Evaluation:

This activity does not change the operation, function or design bases of any structure, system or component important to safety as described in the Updated FSAR. In particular, the undervoltage protection scheme still used relays to detect a degraded voltage condition and to actuate sequencer trip using a two-out-of-two logic. This modification was evaluated under the requirements of 10 CFR 50.59 and did not constitute an unreviewed safety question. The Technical Specification requirements applicable to this modification are not affected. However, the method of satisfying the Technical Specification requirements was affected. While the setpoint values were unchanged, the means of satisfying these setpoint requirements was accomplished by the new definite time delay undervoltage relay. The analysis performed to support this circuit modification confirmed that the new relays, at the original setpoints, provide the necessary degraded voltage protection.

PLANT CHANGE/MODIFICATION 92-057

UNIT : 3
TURN OVER DATE : 11/09/92

HHSI THERMAL RELIEF VALVE

Summary:

This modification to Unit 3 Containment Penetration No. 18 piping consisted of adding a new relief valve for the overpressure protection of this piping. The original design of Containment Penetration No. 18 was to relieve thermal and valve leak-by-overpressure conditions through the use of cross-tied relief valve RV-3-859, located in adjacent Penetration No. 17. This relief scheme was eliminated by isolation of the manual cross-tie valve (3-849A) to alleviate operational problems experienced during routine SI accumulator filling operations.

Safety Evaluation:

This modification was for the installation of a relief valve in the portion of the SI system that was originally overpressure protected by relief valve RV-3-859. The new relief valve was equivalent to the relief valve RV-3-859, and performed the same function as this original valve. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-058

UNIT : 4
TURN OVER DATE : 05/04/93

PROCESS RADIATION MONITORING SYSTEM R-4-11 AND R-4-12 REPLACEMENT

Summary:

This engineering package was issued to replace containment particulate and noble gas radiation monitors R-4-11 (particulate) and R-4-12 (gaseous) and associated displays and controls in control room panel 4QR66 with new ones, which were expected to be more reliable and easier to maintain. R-4-11 and R-4-12 provide a means for monitoring the Unit 4 containment atmosphere for radioactivity released from normal operation, from anticipated transients and from accident conditions. These instruments are part of the Engineering Safety Features Instrumentation. The new equipment performed similar functions to the original equipment, with the exception of the capability to monitor the common plant vent. Since R-14 and RaD-6304 originally monitored the common plant vent, the need for R-4-11 and R-4-12 to monitor the common plant vent was no longer necessary and this function was deleted.

Safety Evaluation:

After the monitor replacements, monitors R-4-11 and R-4-12 continued to perform their safety functions, whereby the high radiation level for the channel initiates closure of the containment purge supply and exhaust duct valves and instrument air bleed valves, and initiates control room ventilation isolation as described in the Updated FSAR. The provision for monitoring the common plant vent using R-4-11 and R-4-12 was no longer necessary, since this function continued to be accomplished by monitors R-14 and RaD-6304. Wide range monitor RaD-6304 was used to satisfy the monitoring requirements of Regulatory Guide 1.97. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-059

UNITS : 3 & 4
TURN OVER DATE : 11/16/92

CONTROL ROOM AIR CONDITIONING AND VENTILATION SYSTEM CONTROL MODIFICATION

Summary:

This engineering package modified the control room Air Conditioning and Ventilation System to allow for the independent operation of the three control room air conditioning trains (i.e., air conditioner and air handler unit). This was accomplished by removing the existing common thermostat, controller and control switches, and providing common independent thermostats, one for each air conditioner units (i.e., compressor/condenser). Also, the air handler motor starters were removed and replaced with fuses for circuit and motor protection. The air handlers may run continuously and their operation only depends on their respective A/C train's power source. In addition, the Firestat sensors were disconnected. These modifications eliminated the potential single failure concerns and associated temporary corrective actions for the control room air conditioning system addressed in the Justification for Continued Operation, as identified in JPE-L-86-113.

Safety Evaluation:

This new design provided electrically independent circuits and components for each air conditioning train, so that, any postulated single failure of a circuit or component would only disable its associated air conditioning train. This modification eliminated existing single failure concerns and did not create any new failure modes that could impact nuclear safety. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-063

UNIT : 3
TURN OVER DATE : 11/11/92

REACTOR COOLANT PUMP 3B MOTOR REFURBISHMENT/UPGRADE

Summary:

As part of the on-going program to improve the reliability and performance of all Reactor Coolant Pumps (RCP) at Turkey Point, this engineering package documented the upgrade of the 3B Reactor Coolant Pump Motor. The originally installed motor was replaced with a spare motor which was refurbished at the Westinghouse Electro-Mechanical Division facility. This standard factory refurbishment consisted of inspection and maintenance activities performed to the existing design specifications. In addition, modifications were conducted, concurrent with the refurbishment, to ensure consistency with the latest RCP technology and to realize additional reliability and availability. Upon completion of the modifications, the motor was assembled, balanced and tested and shipped back to PTN.

Safety Evaluation:

The RCP motor does not perform any safety related function, with the exception of providing sufficient inertia (through its flywheel) to ensure sufficient coastdown of the Reactor Coolant Pump after an RCP trip. The RCP motor modifications did not affect the coastdown characteristics of the motor. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-073

UNIT : 4
TURN OVER DATE : 05/25/93

ADDITION OF REVERSE POWER RELAY AND MAIN GENERATOR PROTECTION MODIFICATIONS

Summary:

This engineering package modified and upgraded the Turkey Point Unit 4 main generator protection schemes. These modifications realigned the existing main generator protection schemes to mitigate the effects of a partial loss of vital DC power on their protection capability. These modifications included the realignment of the vital DC control power supplying these schemes and added a reverse power relay. In addition, this modification upgraded the existing generator protection by regrouping existing main generator primary and backup protective functions, added new relays for out-of-step and 100% ground protection and an emergency control switch to trip the main generator output circuit breakers. Also included as part of this upgrade was the addition of new protection schemes for inadvertent connection, string bus differential and automatic synchronizing. The Institute of Nuclear Power Operations (INPO) Significant Operating Experience Report (SOER) 81-15 made specific recommendations regarding an operating plant's ability to manage and recover from a loss of a vital DC bus. Implementation of these modifications satisfied INPO SOER 81-15 recommendation 1c.

Safety Evaluation:

These modifications to the main generator protection schemes enhanced and upgraded the original non-safety related main generator protection system. No credit is taken for these protection features to accomplish safe shutdown during a design basis accident. The generator protection system remained functionally the same as a result of this modification and the effect of failure of any relay remained unchanged from the original design. Also, the function and performance capability of the auxiliary relays which interface with the safety related diesel generator remained unchanged and unaffected by this modification. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-074

UNIT : 3
TURN OVER DATE : 11/08/92

CORE EXIT THERMOCOUPLE SEAL UPGRADE

Summary:

In order to reduce the potential for primary boundary leakage around the core exit thermocouple (CET) nozzles, this engineering package was implemented to install a new, single piece head port adapter on each CET nozzle, thereby, eliminating the lower Conoseal joint and replacing the upper Conoseal joint with a Grafoil graphite seal. The upper Grafoil seal cartridge was softer and more forgiving than the Conoseal and wil be replaced each time the seal is reassembled. This seal also allowed for a one-joint disassembly for head removal at each outage, which resulted in significant time and radiation exposure savings. The four original core exit thermocouple nozzles on the Unit 3 reactor vessel closure head each have two primary pressure boundary "Conoseal" metal seals that must be disassembled at each refueling outage. The Conoseal installation techniques and surface finish on sealing faces were extremely critical in preventing degradation of the sealing capability for preventing RCS leakage. In recent years, Turkey Point has had several leaks at the Conoseal upon returning to service after an outage. Repair of these leaks has required significant unplanned outage time and lost power generation. The problems with the Conoseal design originally installed were attributed to the difficulty in assembling the seal and degradation of the sealing surfaces which had occurred during the many times they were disassembled and reassembled.

Safety Evaluation:

This modification replaced an existing component with one providing the same function, containing the same basic pressure retaining components, and designed to meet or exceed the original ASME Boiler and Pressure Vessel code requirements. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-075

UNIT	:	4
TURN OVER DATE	:	05/12/93

CORE EXIT THERMOCOUPLE SEAL UPGRADE

Summary:

In order to reduce the potential for primary boundary leakage around the core exit thermocouple (CET) nozzles, this engineering package was implemented to install a new, single piece head port adapter on each CET nozzle, thereby, eliminating the lower Conoseal joint and replacing the upper Conoseal joint with a Grafoil graphite seal. The upper Grafoil seal cartridge was softer and more forgiving than the Conoseal and will be replaced each time the seal is reassembled. This seal allowed for a one-joint disassembly for head removal at each outage, which resulted in significant time and radiation exposure savings. The four original core exit thermocouple nozzles on the Unit 3 reactor vessel closure head each have two primary pressure boundary "Conoseal" metal seals that must be disassembled at each refueling outage. The Conoseal installation techniques and surface finish on sealing faces were extremely critical in preventing degradation of the sealing capability for preventing RCS leakage. In recent years, Turkey Point has had several leaks at the Conoseal upon returning to service after an outage. Repair of these leaks has required significant unplanned outage time and lost power generation. The problems with the Conoseal design originally installed were attributed to the difficulty in assembling the seal and degradation of the sealing surfaces which had occurred during the many times they were disassembled and reassembled.

Safety Evaluation:

This modification replaced an existing component with one providing the same function, containing the same basic pressure retaining components, and designed to meet or exceed the original ASME Boiler and Pressure Vessel code requirements. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-079

UNIT : 4
TURN OVER DATE : 04/30/93

REPAIR AND MODIFICATION OF THE UNIT 4 INTAKE STRUCTURE

Summary:

This engineering package restored the concrete slabs supporting Unit 4 ICW 4A and 4C Pumps and the Screen Wash Pumps to a condition which met the original design bases, and ensured acceptable long-term performance. This was accomplished for the ICW 4A and Screen Wash Bays by removing deteriorated concrete and reinforcing steel, protecting uncovered reinforcing steel from future corrosion, and replacing concrete with material of equal or greater strength. Repairs were implemented using Nonconformance Reports, which were then evaluated and dispositioned by Engineering providing appropriate repairs for each bay.

Safety Evaluation:

Upon completion of the modifications, the structural integrity of the Intake Structure slabs were restored to withstand all applicable loads in accordance with the requirements for Class I structures of Updated FSAR, Appendix 5A, including operating loads of the pumps and associated components, thereby meeting the original design intent of the slab. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-097

UNIT : 4
TURN OVER DATE : 05/12/93

ALTERNATE SAFETY INJECTION THERMAL RELIEF VALVE MODIFICATION

Summary:

This modification consisted of adding a new relief valve in the piping near Unit 4 Penetration No. 18 to provide overpressure protection for this piping. This new relief valve was installed with the inlet line connected directly to the 2-inch alternate safety injection piping and the discharge connected to the existing 3-inch discharge line downstream of relief vale RV-4-382. This new relief valve provided the same overpressure protection function as did RV-4-859, which was part of the original design the for Safety Injection System (SIS). The relief function for thermal and valve leak-by type overpressure conditions at Penetration No. 18 was originally performed by relief valve, RV-4-859, located on adjacent Penetration No. 17. This relief scheme was eliminated through isolation of the manual cross-tie valve, 4-849A, to alleviate relief valve lifting problems experienced during routine SI accumulator filling operations. This engineering package also provided recommendations for Operating Procedures which were revised to eliminate or minimize the Safety Injection System hydraulic transients that may occur during the safety injection accumulator filling operation.

Safety Evaluation:

This modification did not change the operation, function or design bases of any structure, system or component important to safety as described in the Updated FSAR. The components installed under this modification met or exceeded the requirements on the system where they were installed. Also, the affected portion of the SI system was returned to a configuration equivalent to its original design; therefore, no new flow path was created. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-102

UNITS : 3 & 4
TURN OVER DATE : 03/15/93

REPLACEMENT OF RAW WATER STORAGE TANK I (T63A)

Summary:

This engineering package provided the design documentation necessary to replace Raw Water Storage Tank I and to provide appropriate repairs to the damaged foundation. On August 24, 1992, Turkey Point Nuclear site sustained wind damage from Hurricane Andrew. During this event, Raw Water Storage Tank I (T63A), which supplied water to the plant fire protection system and provided potable and service water to the plant, was demolished when the nearby elevated water tank collapsed on top of it.

Safety Evaluation:

The Raw Water Storage Tank I does not perform a safety related function. It is, however, part of the plant fire protection system. The suction nozzle location for the new raw water and service water system remains unchanged on the replacement tank, and thus, the fire protection water reserve capacity is unaffected by the tank replacement. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-108

UNITS : 3 & 4
TURN OVER DATE : 03/19/93

REPLACEMENT OF RAW WATER AND SERVICE WATER SYSTEM DAMAGED BY HURRICANE ANDREW

Summary:

This engineering package provided the design and evaluation required for replacement of the damaged portions of the service water and raw water system. On August 24, 1992, Hurricane Andrew passed over the Turkey Point Power Plant. An elevated water tank located between the fossil plant intake and the nuclear plant intake collapsed due to the storm and damaged equipment located beneath it, including the Raw Water Tank I, portions of the Fire Protection system, raw water booster (service water) pumps, raw water pumps, associated piping, valves, instrumentation, and power supply. The elevated storage tank was eliminated from the system design; however, a diesel engine driven service water pump was added to the system to provide an alternate water source in the event of a loss of electric power to the service water pumps.

Safety Evaluation:

The service water and raw water systems do not provide any safety related functions, however, connections to the Fire Protection system were restored in accordance with the original system design. The changes provided in this PC/M do not alter the functions of the service water or raw water systems, and there was no change to the overall operation of the plant. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-110

UNITS : 3 & 4
TURN OVER DATE : 12/29/92

INSTALLATION OF A DUCT BANK FROM MH 610 TO MH 324

Summary:

This engineering package provided design details for the construction of a duct bank which formed part of the raceway for new non-safety related cables routed from Load Center 3F and 3G to service water pumps P235A, P235B and P235C (the new cables will be installed under PC/M 92-108). On August 24, 1992, Turkey Point Nuclear site sustained wind damage from a Category A hurricane, designated Hurricane Andrew. During this event, service water pumps P235A, P235B and P235C, and the associated power supply were severely damaged, when the nearby elevated water tank collapsed. The power for the non-safety related service water was originally supplied from fossil Units 1 and 2. However, due to the security separation of the fossil and nuclear units, power for the pumps was supplied from the Load Center 3F and 3G.

Safety Evaluation:

The existing manholes MH 324 and MH 610, the duct bank, and the new non-safety related cables pulled through the duct bank to supply power and control functions to service water pumps P235A, P235B and P235C did perform any safety related functions and had no potential for interaction with safety related equipment/systems. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-124

UNITS : 3 & 4
TURN OVER DATE : 03/30/93

OFFSITE RADIO COMMUNICATIONS PROJECT

Summary:

In order to provide a more reliable, permanent installation for the Offsite Radio Communications System, this engineering package provided the design and evaluation for a new permanent system that will remain functional before, during, and after an event similar to Hurricane Andrew. On August 24, 1992 Hurricane Andrew passed over the Turkey Point Power Plant. The original Offsite Radio Communication System design did not remain functional due to the hurricane winds experienced during Hurricane Andrew. The original Offsite Radio Communication System design was not provided with a reliable power source in the event of loss of offsite power (LOOP), and its design did not provide for redundancy or diversity to ensure offsite communications would remain available. Based on the results of the tests, performed throughout the PTN Site, the five radio systems and locations provide reliable and acceptable offsite communications. Three systems are within the project scope of this modification. This design provided diverse and wireless path of communications to local and remote FPL facilities, the emergency operations centers of local counties, and the State and Federal agencies during times of off-normal and emergency events. These systems will remain functional during any foreseeable natural events that are within the design envelope of the plant.

Safety Evaluation:

The function of the new Offsite Radio Communications System is similar to the original system, to provide offsite communications between the Plant and various external and internal organizations in the event of an emergency and loss of normal communications. The new design is intended to provide a more reliable Offsite Radio Communications System capable of withstanding an event similar to Hurricane Andrew. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-163

UNIT : 4
TURN OVER DATE : 05/22/93

REPLACEMENT OF SEAL TABLE FITTINGS AND THIMBLE TUBE LENGTHENING

Summary:

This engineering modification provided for the replacement of the existing seal table fittings with new fittings that contained an integral low pressure seal for refueling. The original fittings were frequently a source of primary leakage during plant startup and operation. The original low pressure refueling seals were difficult to assemble and often leaked.

The new design eliminated the existing guide tube ferrule which was frequently the location of leakage when the seal is configured for plant operation. The replacement design was welded to the guide tube and had a tapered machined sealing surface that replaced the ferrule. The new refueling seal utilized the same sealing technique as the original seal, i.e., compression of an elastomer to form the seal. The difference in the refueling seal was that the new fitting had the seal permanently installed and used an internal nut for compression. This modification also provided for lengthening of thimble tubes to compensate for reduced core inserted thimble end elevations. This condition was associated with new fitting stackup dimensions and previous thimble tube shortening activities.

Safety Evaluation:

The new seal table fittings, thimble tube extensions and replacement guide tubes were similar to the existing hardware and incorporated some improvements. The method of capping abandoned thimble tubes provided an improved and more reliable seal design than the original isolation valve in the event of a thimble tube leak inside the RCS pressure boundary. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-166

UNIT : 4
TURN OVER DATE : 04/15/93

NIS SOURCE RANGE DETECTOR REPLACEMENT

Summary:

The modifications provided by this engineering package cover the replacement of the original BF3 detector model WL-23706 with the new BF3 model NY-10032 in both source range channels for Unit 4. The replacement of the existing BF3 detector model WL-23706 with the new BF3 improved model NY-10032 was required because of the high number of failures experienced with the existing detectors, mainly during refueling outages, and the limited service life of the existing detectors. The replacement detector was an integral cable proportional counter assembly with similar dimensions and parameters as the original detector. The new detector with titanium housing was corrosion resistant and is expected to increase the service life of the detector and reduce the number of detector failures.

Safety Evaluation:

The replacement of the detectors with an improved model did not change the function of the NIS source range channels. Therefore, the system will perform its safety functions as originally designed. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-181

UNIT : 4
TURN OVER DATE : 04/30/93

ELIMINATION OF TURBINE RUNBACK ON DROPPED ROD

Summary:

This engineering package provided the necessary design documentation to remove the turbine runback selector switch HS-4-6686 and associated relays from the Control Room Panel 4C02 to eliminate the activation of a turbine runback on a dropped rod event. The automatic turbine runback feature of Units 3 and 4 is designed to provide protective action in the event of a dropped RCCA or dropped bank. Detection of a dropped RCCA or bank occurs by either a rod-on-bottom signal or by a chnage in neutron flux as seen by the NIS excore power range detectors. The design of the automatic turbine runback on a dropped rod was prone to spurious runbacks (i.e., runbacks not caused by an RCCA drop), because there was no coincidence logic used in the initiation of the runback. Thus, a single failure of an electrical component could cause a turbine runback when it was not needed. Due to the fact that the majority of the spurious runbacks had resulted from failures in the flux rate input to the runback logic, this input was deleted during normal operation. The NIS switch position was only used for short time intervals while performing periodic maintenance or tests. This modification to the Turbine Runback System was analyzed in Appendix 14C of the updated FSAR. This evaluation concluded that deletion of the flux rate portion of the Turbine Runback System is acceptable. The reactor can be maintained in automatic rod control, since auto rod withdrawal had been previously eliminated.

Safety Evaluation:

The RCCA Drop analysis in the FSAR is currently analyzed with the protective action of turbine runback. The Dropped RCCA transient assuming no turbine runback was analyzed by Westinghouse using a detailed digital simulation of the Turkey Point Plant. The results of the analysis confirmed that the departure from nucleate boiling (DNBR) remains above the limiting value for both standard and optimized fuel types. Thus, it was concluded that eliminating turbine runback following a dropped rod event would not have an adverse impact on plant safety. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 93-009

UNIT : 4
TURN OVER DATE : 05/07/93

INSTALLATION OF JIB CRANE IN THE UNIT 4 CONTAINMENT BUILDING AT ELEVATION 58'-0"

Summary:

This engineering package provided the necessary engineering requirements and details for the procurement and installation of a jib crane in the Unit 4 Containment Building. Originally, the polar crane was used to move relatively small loads between two containment building elevations. Tool boxes, small equipment, etc., were staged and removed from the 58'-0" elevation during early and latter parts of a refueling outage. At these times, the use of the polar crane was limited to handling large containment loads. For lighter loads, personnel were used to move the loads via the stairs or the containment elevator. This engineering package removed these ineffective and potentially unsafe load transfer activities by installing a jib crane. The jib crane is capable of hoisting loads up to 1,500 pounds. The crane is 10 feet high, with an eight foot rotating boom. The jib crane can be used during Modes 5, 6 and defueled. The jib crane will not be used in Modes 1, 2, 3 or 4 due to potential for a load drop accident scenario causing unsafe plant conditions.

Safety Evaluation:

The jib crane, installed along the west side of the equipment hatch was limited to 1,500 pounds to avoid heavy load requirements which would result in substantial structural modifications due to large factors of safety. The jib crane was equipped with boom rotation stops to prevent striking the containment liner plate with the end of the boom. The jib crane was not a single failure proof device. As a result the jib crane operation will be limited to Modes 5, 6 or defueled in order to preclude the potential for an inadvertent load drop scenario. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 93-020

UNIT : 4
TURN OVER DATE : 05/21/93

REACTOR COOLANT PUMP 4A MOTOR REFURBISHMENT/UPGRADE

Summary:

As part of the on-going program to improve the reliability and performance of all Reactor Coolant Pumps (RCP) at Turkey Point, this engineering package documented the upgrade of a spare motor to be installed in the 4A RCP slot. The original motor was replaced with the rotated spare motor, which was refurbished at the Westinghouse Electro-Mechanical Division facility. This standard factory refurbishment consisted of inspection and maintenance activities performed to the existing design specifications. In addition, modifications were conducted, concurrent with the refurbishment, to insure consistency with the latest RCP technology and to realize additional reliability and availability. Upon completion of the modifications, the motor was assembled, balanced and tested and shipped back to PTN.

Safety Evaluation:

The RCP motor does not perform any safety related function, with the exception of providing sufficient inertia (through its flywheel) to ensure sufficient coastdown of the Reactor Coolant Pump after an RCP trip. The RCP motor modifications did not affect the coastdown characteristics of the motor. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION JPE-M-86-011
REVISION 0

UNIT : 4
TURNOVER DATE : 03/15/93

SAFETY EVALUATION FOR CPWOs 86-017 AND 86-018
UNIT 4 REPLACEMENT OF NORMAL & EMERGENCY CONTAINMENT COOLER DRIP PANS

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 86-017 and PC/M 86-018, whose implementation activities were completed and turned over to the plant by March 15, 1993. The purpose of these PC/Ms was to fabricate and replace the existing drip pans for the Unit 4 Normal and Emergency Containment Coolers inside containment. The original galvanized steel drip pans collected condensate from the cooling coils and were in a corroded condition. These galvanized drip pans were replaced with stainless steel pans of the same thickness, which did not require any additional supports or restraints due to the equivalency of weights of both assemblies. All rework duplicated the existing drip pan design with the exception of material. Similarly, all carbon steel fittings were replaced with equivalent stainless steel fittings.

Safety Evaluation:

All replacement drip pans duplicated the existing drip pan design with the exception of material. The removal and reinstallation of the Normal and Emergency Containment Cooler drip pans did not affect any other system nor require any other component to be taken out of service. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The plant changes in hardware did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this safety evaluation.

SAFETY EVALUATION JPE-M-86-033
REVISION 0

UNITS : 3 & 4
TURNOVER DATE : 05/13/93

SAFETY EVALAUTION FOR CPWO 86-086
RELOCATION OF EMERGENCY DIESEL GENERATOR COOLING SYSTEM DRAIN VALVES 293A AND 293B

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 86-086, whose implementation was completed and turned over to the plant by May 13, 1993. The purpose of this PC/M was to relocate Emergency Diesel Generator (EDG) A & B cooling system drain valves 293A and 293B from their existing location outside the vital area barrier to within the barrier confines for the EDGs. In addition, valves 292A and 292B were located on the same drain lines from the diesel radiator shells, but were located on inside the vital area barriers. Under normal operating conditions, all valves are normally closed with 293A and 293B locked closed.

Safety Evaluation:

The relocation of the drain lines did not affect the intended function of the EDG radiator shell drain lines or associated isolation valves. A seismic evaluation performed on the revised configuration of the radiator drain line with isolation valves installed demonstrated that this revised configuration could withstand a design basis seismic event. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The plant changes in hardware did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this safety evaluation.

SAFETY EVALUATION JPE-M-86-067
REVISION 0

UNITS : 3 & 4
TURNOVER DATE : 03/29/93

SAFETY EVALUATION FOR CPWOs 86-163 PASS CHLORIDE REAGENT AND CALIBRATION STANDARD PUMPS SUBSTITUTION

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 86-163, whose implementation activities were completed and turned over to the plant by March 29, 1993. The purpose of this PC/M was to substitute two existing Post Accident Sampling System (PASS) chloride reagent and calibration standard positive displacement metering pumps with pumps of a different type. The original positive displacement pumps design discharge pressure range was 20 and 30 psig, respectively; while, the system pressure range was approximately 50 psig. This excessively high pressure differential was considered to directly contribute to the high failure rate of the original type of pumps. The replacement of the original pumps with diaphragm-type pumps was considered to be an appropriate substitution for this system. After replacement pump testing was performed to verify that no leakage was present.

Safety Evaluation:

The replacement of the original pumps with diaphragm-type pumps was considered to be an appropriate substitution for this system. The work performed under this PC/M did not affect any plant features necessary to assure the integrity of the reactor coolant pressure boundary, nor did it hamper the capability to shutdown the reactor and maintain in safe shutdown condition following design basis events. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The plant changes in hardware did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this safety evaluation.

SAFETY EVALUATION JPES-PTP-86-433E
REVISION 0

UNIT : 4
TURNOVER DATE : 03/23/93

SAFETY EVALUATION FOR CPWO 86-035
UNIT 4 VALVE POSITIONER REPLACEMENT FOR PCV-4-455 A & B

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 86-035, whose implementation activities were completed and turned over to the plant by March 23, 1993. The purpose of this PC/M was replacement of the valve positioner on pressurizer spray valves PCV-4-455A and PCV-4-455B. The original Bailey positioner model was no longer available and a suitable replacement approved for nuclear service was selected, i.e., Barton Conoflow positioner. Replacing the original positioner with the Barton Conoflow positioner enhanced the reliability of the pressure spray valve operation, since the replacement positioner had better documented qualifications than the original positioner, which was supplied as an accessory on the Copes-Vulcan valves. In addition, since the Copes Vulcan valves had been relocated outside of the pressurizer cubicle with its normally high radiation levels, the expected dose to replacement positioner would be substantially reduced.

Safety Evaluation:

The conclusion of the safety evaluation was that the positioner replacement was considered to enhance the reliability of the pressure spray valve operation, since the replacement positioner had better documented qualifications and was located in a lower radiation environment. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The plant changes in hardware did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this safety evaluation.

SAFETY EVALUATION JPES-E-87-384
REVISION 0

UNIT : 4
TURNOVER DATE : 03/29/93

SAFETY EVALUATION FOR CPWOs 87-060 AND 87-061 PRMS DRAWER REPLACEMENT

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 87-061, whose implementation was completed and turned over to the plant by March 29, 1993. The purpose of this PC/M was to replace several PRMS drawers which required maintenance. Each PRMS drawer contained the electronics and hardware to power its associated radiation detector, to process the signal, and to provide digital and analog outputs. Due to schedular problems, the original equipment could not be supplied in a timely fashion and the vendor offered an upgraded model of drawer. This seismically qualified upgraded PRMS drawer was installed as a replacement for existing drawers in Unit 4 PRMS channels R-11, R-12, R-15, R-17A, R-17B, and R-19. The replacement drawers were equivalent in form, fit and function to the original equipment. Performance specifications for the new drawers indicated that parameters, such as, accuracy, response time, and repeatability were equivalent to the original drawers.

Safety Evaluation:

The upgraded PRMS drawer replacement did not change the system functional design basis or the system configuration. In addition, the replacement drawers were equivalent in form, fit and function to the original equipment. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The plant changes in equivalent hardware did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEEJ-88-042
REVISION 2

UNIT : 4
APPROVAL DATE : 04/08/93

DE-ENERGIZATION OF UNIT 4 4160 VOLT SAFETY RELATED BUSSES

Summary:

This evaluation was developed to establish the requirements and restrictions which must be placed on the operation of Units 3 and 4 and their equipment when a Unit 4 4160 volt bus was de-energized and Train A and B load centers were cross-connected. Also examined were technical and licensing concerns associated with de-energizing safety related equipment and removing on EDG from service concurrent with a Unit 4 4160 volt bus de-energization. The de-energization of a Unit 4 4160 volt safety related bus, with Unit 4 in cold or refueling shutdown (Modes 5 or 6) or de-fueled and Unit 3 at power operation (Mode 1) or below, is sometimes necessary to allow for periodic maintenance, testing, or design modifications of the 4160 volt switchgear. De-energization of a 4160 volt bus would cause de-energization of the 480 volt load centers and motor control centers powered from that bus, if any, and a loss of power to equipment which may be required to maintain cold/refueling shutdown, perform outage related activities, or support safe shutdown and accident mitigation on the opposite Unit. This condition was alleviated by closing the tie-breakers between opposite train 480 volt load centers, while one 4160 volt bus was de-energized or by ensuring that alternate equipment was available.

Safety Evaluation:

This safety evaluation addressed the technical and licensing requirements for the de-energization of each Unit 4 4160 volt bus and concluded that the proposed plant configuration and mode of operation was bounded by the Technical Specifications and did not change the analysis of accidents addressed in the FSAR or the results and conclusions of any previous safety evaluations. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMJ-88-090
REVISION 0

UNIT : 3
TURNOVER DATE : 08/11/92

SAFETY EVALUATION OF THE DELETION OF FIRE HOSE STATIONS IN THE RADWASTE BUILDING

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 88-603, whose implementation activities were completed and turned over to the plant by August 11, 1992. This PC/M was developed to correct drawing descrepancies identified on Operating Diagram 5610-T-E-4072, Sheet 1. A field verification walkdown of the fire protection system and a review of 5610-T-E-4072, Sheet 1 revealed several drawing discrepancies. This drawing showed to fire hose stations in the Radwaste Building as part of the fire protection system. However, field walkdowns confirmed that these two hose stations were actually part of the non-safety related service water system and not the fire protection system. This drawing was corrected to match the existing field confirguration. This safety evaluation verified the ability of the existing fire protection system to meet licensing basis criteria from the Updated FSAR.

Safety Evaluation:

The investigation confirmed that the Radwaste Building was not required to be covered under the fire protection program to meet NRC requirements, and therefore hose stations were not required in that building. Therefore, the Updated FSAR and engineering drawings may be updated accordingly without affecting the validity of the plant fire protection program and the overall safe operation of the plant. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The document changes did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this PC/M and safety evaluation.

SAFETY EVALUATION JPN-PTN-SEEJ-89-085
REVISION 3

UNIT : 3
APPROVAL DATE : 10/08/92

DE-ENERGIZATION OF UNIT 3 4160 VOLT SAFETY RELATED BUSSES

Summary:

This evaluation was developed to establish the requirements and restrictions which must be placed on the operation of Units 3 and 4 and their equipment when a Unit 3 4160 volt bus was de-energized and Train A and B load centers were cross-connected. Also examined were technical and licensing concerns associated with de-energizing safety related equipment and removing an EDG from service concurrent with a Unit 3 4160 volt bus de-energization. The de-energization of a Unit 3 4160 volt safety related bus, with Unit 3 in cold or refueling shutdown (Modes 5 or 6) or de-fueled and Unit 4 at power operation (Mode 1 or below) is sometimes necessary to allow for periodic maintenance, testing, or design modifications. De-energization of a 4160 volt bus would cause de-energization of the 480 volt load centers and motor control centers powered from that bus, and a loss of power to equipment which may be required to maintain cold/refueling shutdown, perform outage related activities, or support accident mitigation on the opposite unit. This condition was alleviated by closing the tie-breakers between opposite train 480 volt load centers, while one 4160 volt bus was de-energized or by ensuring that alternate equipment was available.

Safety Evaluation:

This safety evaluation addressed the technical and licensing requirements for the de-energization of each Unit 3 4160 volt bus. It concluded that the proposed plant configuration and mode of operation was bounded by the Technical Specifications and did not change the analysis for accidents addressed in the FSAR or the results and conclusions of any previous safety evaluations. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENP-92-001
REVISIONS 1 & 2

UNITS	:		3 & 4
APPROVAL DATES	:	Rev.1	08/06/92
	:	Rev.2	03/11/93

THE CONDUCT OF INTEGRATED SAFEGUARDS TESTING ON A SHUTDOWN UNIT WITH THE OPPOSITE UNIT AT POWER

Summary:

The purpose of this evaluation was to identify and resolve technical and licensing concerns associated with the performance of integrated safeguards testing on one unit with the opposite unit at power. This evaluation established the acceptability of performing train-by-train integrated testing on the shutdown unit with the opposite unit at power or any other mode of operation, while confirming that all technical specification requirements were met. The principal objectives of the revised test procedure as evaluated in this safety evaluation was to satisfy Technical Specification surveillance and test requirements for the onsite emergency power system and safeguards equipment on a train-by-train basis, while allowing continued, uninterrupted power operation of the non-test unit without placing either unit in an unanalyzed or unsafe condition.

Revision 1 of this evaluation incorporated a review of the current changes to the IST procedure and incorporated minor plant comments. Revision 2 of this evaluation incorporated changes to the EDG loading charts, which address limitations experienced with the Unit 3A EDG during surveillance testing and provided additional acceptance criteria. All other portions of this safety evaluation remained unchanged.

Safety Evaluation:

Based on the requirements, restrictions and precautions specified and discussed in this safety evaluation and Technical Specifications, the performance of the proposed revised integrated safeguards testing in accordance with the revised plant procedures, 3/4-OSP-203.1 and 3/4-OSP-203.2, did not have any adverse effect on plant safety or plant operations. The actions and plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions and changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEEP-92-008
REVISION 0

UNITS : 3 & 4
TURNOVER DATES : 92-030 01/07/93
92-032 05/21/93

SAFETY EVALUATION FOR LOAD CENTER AND RELAY SETTING CHANGES

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 92-030 and PC/M 92-032, whose implementation activities were completed and turned over to the plant by January 7, 1993 and May 21, 1993, respectively. These PC/Ms were developed to resetting one overcurrent relay per unit and provide overcurrent circuit breaker settings for safety related load center breakers. The overcurrent relay which was reset on each unit was for the Train "A" 4160 VAC switchgear circuit breaker feed from the adjacent unit's start-up transformer. This alternate electrical feed provided each unit with a second source of offsite emergency power which was capable of supporting the loads necessary for achieving and maintaining safe shutdown. Providing the overcurrent circuit breaker settings for the safety related load center breakers was intended to document the engineering specified settings and incorporated these settings into the controlled drawing system. This safety evaluation established the basis for any relay setting changes that were necessary.

Safety Evaluation:

Overcurrent relay and circuit breaker settings served to protect equipment during electrical fault and abnormal overload conditions. The overcurrent device settings were based on engineering calculations which ensured electrical protection and coordination. These changes did not add any new component or change the function, operation, and design basis of any existing equipment described in the SAR. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The document changes did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in these PC/Ms and safety evaluation.

SAFETY EVALUATION JPN-PTN-SEFJ-92-012
REVISIONS 1 & 3

UNITS	:		3 & 4
APPROVAL DATES	:	Rev.1	08/20/92
		Rev.3	04/16/93

EVALUATION OF IMPACT OF ACCUMULATOR DISCHARGE TEST ON FUEL AND REACTOR INTERNALS

Summary:

This safety evaluation documented the acceptability of performing the accumulator discharge test with fuel and upper intenals in the reactor vessel. The test allowed the full stroke exercise of the accumulator discharge check valves. It was performed by discharging the accumulator water volume into the RCS at a predetermined pressure that was sufficient to fully open the downstream check valves. This test was previously done on Unit 3 without fuel in the reactor vessel. In this evaluation, Nuclear Fuels investigated any adverse effects on the fuel and reactor vessel internals which could occur as a result of this test.

Revision 1 to this evaluation examined the performance of the test with the upper internals in the reactor vessel and addressed industry experience with this test, such as, the Wolf Creek Plant contamination incident of 1988. Revision 3 provided tolerances and clarification of the 50 second requirements to close the accumulator isolation valve after the initiation of the test.

Safety Evaluation:

The proposed configuration was a normal plant evolution in Mode 6, in which mode the test was performed. The effects of the accumulator discharge test on the fuel and the reactor internals were bounded by normal plant evolutions. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEFJ-92-015
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 09/25/92

EVALUATION OF ACCUMULATOR DISCHARGE TEST WITH REACTOR VESSEL HEAD INSTALLED

Summary:

This safety evaluation documented the acceptability of performing an accumulator discharge test with fuel and upper internals in the reactor vessel and the reactor vessel head installed. The test allowed the full stroke exercise of the accumulator discharge check valves. It was performed by discharging accumulator water volume into the RCS at a predetermined pressure that was sufficient to fully open the downstream check valves. This test was previously done on Unit 3 without fuel in the reactor vessel and the reactor vessel head removed. The impact on fuel and reactor internals of the increased flow experienced in the reactor vessel during the opening of the accumulator isolation valve, and the potential for release of nitrogen into the reactor vessel had been evaluated previously in another safety evaluation.

This safety evaluation focused on additional potential issues resulting from the reactor vessel head being installed. These areas of concern were identified as follows: (1) RCS pressurization at low temperature and the impact on the circumferential weld of the reactor vessel; (2) the flow increase experienced in the pressurizer and the impact on the pressurizer heaters; (3) over-filling of the pressurizer and spilling into the Pressurizer Relief Tank (PRT); and (4) the impact on accumulator thermal stresses.

Safety Evaluation:

The proposed configuration was a normal plant evolution in Mode 6, in which mode the test was performed. The effects of the accumulator discharge test on the fuel and the reactor internals were bounded by normal plant evolutions. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECS-92-018
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 08/04/92

TEMPORARY LEAD SHIELDING INSTALLATION SPECIFICATION SPEC-C-003

Summary:

This engineering evaluation examined Specification SPEC-C-003, "Temporary Lead Shielding Installation Specification", which provided engineering guidance and requirements for the installation of temporary lead shielding at Turkey Point Units 3 and 4. Lead shielding could be in the form of blankets, sheets or bricks which could be configured to form temporary lead shielding barriers. These barriers could be supported from permanent or temporary plant structures, or could be applied directly to piping systems. The specification prohibited the use of these barriers in Modes 1, 2, 3 or 4, and allowed their use in Modes 5, 6 or defueled, only if the specific set of implementation instructions accompanying each shielding barrier allowed it. The intent of the specification was to present a convenient set of temporary plant configurations which had been assessed by engineering for impact on nuclear safety. This was done to avoid performing repetitive engineering evaluations for the installation of frequently used lead shielding barriers.

Safety Evaluation:

Temporary lead shielding barriers covered under the scope of the specification did not perform safety related functions, nor did they alter plant operations, design bases or technical specifications. Their installation was considered a temporary change to the facility which was evaluated to satisfy plant licensing requirements. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECP-92-021
REVISION 0

UNIT : 4
APPROVAL DATE : 08/11/92

UNIT 4 TWENTIETH YEAR CONTAINMENT TENDON SURVEILLANCE

Summary:

The purpose of this safety evaluation was to address construction and surveillance activities associated with the Unit 4 twentieth year containment structure tendon surveillance. The tendon surveillance program is an in-service physical inspection of the concrete containment post-tensioning system, which satisfied plant Technical Specification requirements for the twentieth year surveillance. This surveillance program is a systematic means of assessing the continued performance of the containment post-tensioning system. Technical Specification Section 4.6.1.6.1 required that three dome, five hoop (horizontal), and four vertical tendons be selected for the twentieth year surveillance based on a random and representative selection process. Also, other inspection activities were performed for data collection purposes, but were not required to satisfy Technical Specifications.

Safety Evaluation:

The performance of this tendon surveillance did not compromise the containment structural integrity, because the conditions for testing, as described in Updated FSAR Section 5.1.7.4 and Technical Specification Sections 4.6.1.6.1 and 4.6.1.6.2 were maintained. Similarly, this activity did not create any spatial or functional adverse interaction with any structure, system or component important to safety or safe plant operation. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENP-92-032
REVISIONS 0-2

UNITS	:		3 & 4
APPROVAL DATES	:	Rev.0	09/03/92
	:	Rev.1	09/04/92
	:	Rev.2	10/02/92

THE DEMOLITION OF THE TURKEY POINT FOSSIL UNIT 1 CHIMNEY

Summary:

The purpose of this safety evaluation was to analyze the effect on the nuclear units during a controlled demolition of the Turkey Point Fossil Unit 1 chimney. Hurricane Andrew hit south Florida causing damage to equipment and structures on both the nuclear and fossil units. Selected site damage that was incurred included visible structural damage to the Fossil Unit 1 chimney. Although the Unit 1 chimney remained standing, the extensive nature of the damage raised concerns that the chimney may not have survived another high wind event. The chimney could have fallen in an uncontrolled manner and represented a potential hazard to personnel and the Turkey Point Nuclear Units. Conventional dismantling of the chimney would taken several months and the required technique itself represented a personnel safety hazard. Therefore, the Unit 1 chimney was raised in a controlled manner using precision demolition to cause it to fall in a safe and predictable direction. Based on a detailed inspection, the Unit 2 chimney, which is the closest chimney to the nuclear units, did not suffer any significant structural damage.

Revision 1 clarified the limitations on wind speed and direction applicable to the demolition plan. Revision 2 of the safety evaluation was issued to provide a final report, as described in attachments to the safety evaluation.

Safety Evaluation:

This activity has been evaluated and equipment important to safety will remain functional during and following the felling of the Unit 1 chimney. No adverse interactions involving safety related equipment would be created by the demolition of the Unit 1 chimney. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-92-033
REVISION 0

UNIT : 3
APPROVAL DATE : 08/13/92

FREEZE SEAL EVALUATION FOR REPLACEMENT OF VALVES 3-777, 3-834 AND 3-833

Summary:

This safety evaluation addressed the use of freeze seals, temporary supports, and a seismic evaluation while replacing the Component Cooling Water (CCW) valves 3-777, 3-834 and 3-833. CCW from the non-regenerative heat exchanger bypass throttle valve 3-834, CCW supply to non-regenerative heat exchanger isolation valve 3-777, and TCV-3-144 inlet isolation valve 3-833 had all deteriorated to the point where they could not perform their normal functions due to excessive seat leakage. The maintenance was performed in two phases. The first phase, performed in Modes 5 and 6 during the Unit 3 refueling outage, replaced valves 3-777 and 3-834, which required closing valve 3-781 and establishing freeze seals to isolate the work area. The second phase of maintenance replaced valve 3-833, using valves 3-777, 3-834 and 3-780 as boundaries. This phase of the maintenance could have been performed any time during the refueling outage after the completion of the first phase but prior to returning the non-regenerative heat exchanger to service.

Safety Evaluation:

Reduced water inventory operations are sensitive to a loss of decay heat removal capability. However, administrative controls on the freeze seal operation and operator actions precluded the loss of decay heat removal capability during the maintenance activity any adverse interactions with equipment important to safety. Additionally, analysis was performed on the piping and pipe supports to ensure that acceptable loadings were not exceeded any time during the maintenance. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENP-92-033
REVISIONS 0 & 1

UNITS	:		3 & 4
APPROVAL DATES	:	Rev.0	09/19/92
	:	Rev.1	09/24/92

SAFETY EVALUATION RELATED TO THE TURKEY POINT FOSSIL UNIT 2 CHIMNEY

Summary:

The purpose of this safety evaluation was to analyze the effect on safety of operating the nuclear units with the Unit 2 chimney in its current post-hurricane condition. Hurricane Andrew hit south Florida causing damage to equipment and structures on both the nuclear and fossil units. Specific site damage that was incurred included minor vertical and horizontal cracking to the Unit 2 chimney, which stands next to the nuclear units. In most cases the cracks were hairline cracks with little or no spalling or signs of distress within the lower 150 feet. An analysis conducted by Failure Analysis Associates (FaAA) conservatively modeled the cracks and showed that the stack would not fail under the original design load of 55 psf (approximately equivalent to 145 mph wind) or following a 0.15g seismic event. The results of this evaluation were corroborated by a second independent evaluation.

Revision 1 to this safety evaluation incorporated the results of additional structural analyses, which showed structural margins for the Unit 2 chimney of 55% for a 55 psf wind load, 40% for a 145 mph FSAR wind load, and 25% for a 225 mph FSAR tornado wind load. These analyses demonstrated the ability of the Unit 2 chimney to withstand the design basis natural phenomenon (hurricane, tornado, and seismic) without interacting adversely with the nuclear units.

Safety Evaluation:

Although the likelihood of a Unit 2 chimney failure resulting in damage to equipment important to safety is a low-probability event, this remote possibility was evaluated to determine the consequences of such an extraordinary event. This evaluation concluded that even the worst case scenarios of equipment damage could be accommodated with core damage and that current plant procedures are in effect to cope with equipment damage events. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or changes, identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMJ-92-034
REVISIONS 0-2

UNITS	:		3 & 4
APPROVAL DATES	:	Rev.0	09/24/92
	:	Rev.1	09/25/92
	:	Rev.2	09/30/92

INTERIM FIRE PROTECTION SYSTEM CONFIGURATION TO SUPPORT UNIT 4 STARTUP

Summary:

The purpose of this safety evaluation was to identify the Fire Water Supply System licensing and design basis requirements and determine what system configuration requirements were needed for Unit 4 startup following Hurricane Andrew. High winds associated with Hurricane Andrew caused the Turkey Point (PTN) Raw Water System high tower to collapse. As a result of the collapsed high tower, portions of the PTN Fire Water Supply System were damaged, including the electric driven fire pump, both fire water jocket pumps, and portions of the fire protection piping system. Although several plant modifications were under development to restore the Fire Protection facilities, these modifications were not fully implemented in time to support startup of Turkey Point Unit 4. This evaluation examined the interim Fire Protection System configuration which was evaluated against the system operability requirements specified in the Turkey Point Technical Specifications and the updated FSAR. The necessity for supplemental fire protection equipment to meet system design requirements was also determined.

Revision 1 of the evaluation provided additional information regarding the performance capabilities of the screen wash pumps. Revision 2 of the evaluation clarified the operational requirements of the jockey pumps to support unit startup.

Safety Evaluation:

As discussed in this safety evaluation, the interim Fire Water Supply System utilizing the Raw Water Storage Tank II with the permanent electrical and diesel driven fire pumps and the intake canal with the screen wash pumps was capable of delivering the required fire water flow and remained capable of mitigating the effects of a fire. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECP-92-038
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 11/27/92

SAFETY EVALUATION RELATED TO THE TURKEY POINT FOSSIL UNITS 1 AND 2 CHIMNEY CONSTRUCTION ACTIVITIES

Summary:

The purpose of this safety evaluation was to address the construction activities associated with the erection of a new chimney for Turkey Point Fossil Unit 1 and the reinforcement of the Fossil Unit 2 chimney which were damaged during Hurricane Andrew on August 24, 1992. Specific site damage that was incurred included visible structural damage to the Turkey Point Fossil Unit 1 chimney and minor cracking to the Unit 2 chimney. The damage to the Unit 1 chimney was sufficiently severe to require its demolition. The post-hurricane condition of the Unit 2 chimney was evaluated in another safety evaluation. This evaluation concluded that the chimney had sufficient remaining capacity to withstand the Turkey Point Updated FSAR loads for Class I structures without adversely interacting with the nuclear units. However, due to the long term corrosion problems a new sheath would be constructed around the Unit 2 chimney. The scope of this evaluation was limited to the erection of the Unit 1 chimney and preparations for the reinforcement of the Unit 2 chimney up to, but not including, placement of concrete.

Safety Evaluation:

The construction activities associated with the new Unit 1 chimney and reinforcement of the Unit 2 chimney up to, but not including, Unit 2 concrete placement were reviewed. All equipment and materials were confined to the Units 1 and 2 side of the site and would not affect the nuclear units. The new Unit 1 chimney was analyzed to show that it wwould be able to withstand the wind and seismic loads defined in the FSAR for Class I structures without interacting with the nuclear units. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECP-92-040
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 12/08/92

SAFETY EVALUATION RELATED TO THE NEW TURKEY POINT FOSSIL UNIT 1 CHIMNEY AND UNIT 2 CHIMNEY REINFORCEMENT

Summary:

The purpose of this safety evaluation was to document the design criteria which were used in the design of a new Unit 1 chimney and the reinforcement of the original Unit 2 chimney. The original fossil chimney's were damaged during Hurricane Andrew on August 24, 1992. The criteria used ensured that the new and repaired chimneys could withstand the loads defined in the Turkey Point Updated FSAR for Class I structures without interacting with the nuclear units. This evaluation addressed the potential effects of the Unit 2 concrete placement on the safe operation of the nuclear units, since failure of the chimneys and/or construction accidents could potentially affect nuclear safety related equipment.

Safety Evaluation:

The criteria used in the design of the new Unit 1 chimney and the reinforcement of the Unit 2 chimney ensured compliance with all existing building codes, and also ensured that there was no potential for interaction with the nuclear units under the wind and seismic loads defined in the Updated FSAR for Class I structures. The design of the new and reinforced chimneys was verified by an independent consultant (Failure Analysis Associates). The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENS-92-044
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 09/30/92

MANUAL OVERRIDE OF MOV-*-626 DURING RCP SEAL FAILURE

Summary:

This safety evaluation examined the effects of manually overriding the automatic operation of MOV-*-626 (by opening/verifying open the valve) following an reactor coolant pump (RCP) seal failure and using an operator dedicated to restoring electric power to the MOV when required. MOV-*-626 is the CCW return isolation valve common to all the RCP thermal barrier heat exchangers. MOV-*-626 is part of the containment isolation scheme for contaiment Penetration No. 43. A Westinghouse bullentin described how the failure of a No. 1 RCP seal could result in a loss of CCW to all RCP thermal barrier heat exchangers due to the automatic closure of MOV-*-626. This would result in a loss of CCW cooling to all RCP thermal barrier heat exchangers, which potentially leads to the failure of the unaffected RCP seals. This safety evaluation redefined the design basis for containment Penetrations No. 3, 4 and 43 to allow the closed system inside containment to be one of the required barriers.

Safety Evaluation:

The redefined design bases for containment Penetrations No. 3, 4, and 43 satisfy the two barrier criterion for containment isolation and were successfully evaluated against the Updated FSAR single active failure criterion. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENS-92-045
REVISION 0

UNIT : 3
APPROVAL DATE : 08/21/92

FREEZE SEAL INSTALLATION ON THE HHSI ALTERNATE HOT LEG INJECTION CROSS-TIE PIPING

Summary:

This evaluation examined the installation of a freeze plug on the alternate hot leg injection cross-tie header for the performance of flow testing of the High Head Safety Injection (HHSI) pumps. This testing was performed in a mode when the safety injection system was not required to be operable by technical specifications.

Normal maintenance or testing performed on a system not required to be operable by the technical specifications does not generally require evaluation under the provisions of 10 CFR 50.59. However, site policy governing evolutions for the use of freeze seals was under development, and it was considered prudent at the time to evaluate the piping configuration against the criteria of 10 CFR 50.59.

Safety Evaluation:

The freeze seal was installed on the HHSI alternate hot leg injection cross-tie piping which was not required to be operable per the Technical Specifications. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-92-052
REVISION 0

UNIT : 3
APPROVAL DATE : 10/15/92

SAFETY EVALUATION FOR ICW VALVE REPLACEMENT

Summary:

This safety evaluation covered isolation, removal, and replacement of ICW header cross-connect valves 3-50-307 and 3-50-350. The purpose of this safety evaluation was to assess all potential safety concerns associated with activities for the replacement of these valves. The replacement work was performed with Unit 3 in Mode 6 or with the reactor defueled and all the spent fuel stored in the spent fuel pool (SFP). The valves were replaced due to excessive seal leakage, making isolation of the ICW headers during maintenance crawl-through inspections difficult.

Safety Evaluation:

The ICW configurations that were established during the valve replacement maintenance were analyzed to ensure that the operable portions of the ICW system remained seismically qualified. The ability of the ICW system to support Residual Heat Removal (LHSI) and SFP cooling during Mode 6, or SFP cooling with the reactor defueled and all fuel in the SFP was not adversely impacted by the valve replacement activities. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECS-92-056
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 09/22/92

INSTALLATION OF COMMUNICATION ANTENNAS

Summary:

This evaluation addressed the acceptability of installing two fiberglass whip antennas on the control room roof under Temporary System Alteration (TSA) 3-92-1-23. The antennas were attached to the missile barrier separating the computer room HVAC units. As a result of Hurricane Andrew, offsite communications were interrupted due to loss of all communication paths from the site due to equipment damage. In order to increase the capability of the offsite communications system and significantly increase the probability of maintaining offsite communications paths during an event similar to Andrew, a VHF and UHF radio system was installed. This safety evaluation addressed the mounting of the antennas and their potential interaction with safety related structures, systems and equipment because of their location and attachment to the control room missile barrier.

Safety Evaluation:

This safety evaluation concluded that two antennas could be installed on the subject missile barrier provided that all of the requirements stipulated within this evaluation were followed. The evaluation also concluded that this activity will have no adverse impact on plant operations, and will not compromise the licensing bases for Turkey Point. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SENS-92-059
REVISION 0

UNIT : 3
APPROVAL DATE : 10/08/92

UNIT 3 REFUELING OUTAGE CONTINGENCY PLAN FOR EMERGENCY POWER TO THE SFP PUMPS

Summary:

This safety evaluation provided a basis for a contingency plan to provide a source of emergency (back-up) power to the Unit 3 spent fuel pool (SFP) cooling pump motor transfer switch in the unlikely event of loss of normal power from Load Center 3C. The SFP cooling system did not include emergency power as a design requirement, and a SFP boiling analysis demonstrated that offsite doses will remain well within 10 CFR 100 limits. The contingency plan evaluated in this safety evaluation required that a cable of sufficient length be installed (only upon a loss of the normal power supply) between the SFP motor transfer switch and breaker 42116 in cubicle 4E of MCC 4H. The interconnecting cable is for use only during the Unit 3 refueling outages.

Safety Evaluation:

The installation of a temporary cable has been evaluated electrically and seismically and will not adversely affect the SFP cooling system and adds reliability to the system during Unit 3 refueling outages. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-92-060
REVISION 0

UNIT : 3
APPROVAL DATE : 10/09/92

INSTALLATION AND USE OF AN ABB/CE RCCA INSPECTION STATION AT TURKEY POINT

Summary:

This safety evaluation evaluated the consequences of installation of an ABB/CE rod cluster control assembly (RCCA) inspection device at Turkey Point. The inspection device was installed on top of the spent fuel storage racks. This evaluation included the effect of the inspection stand, and the RCCA while in the stand, on the racks only. It did not consider the process of RCCA removal, storage, evaluation, subsequent RCCA disposal or re-insertion in the fuel.

Safety Evaluation:

The potential safety issues associated with installing this equipment were enveloped by postulated accidents previously evaluated in the Updated FSAR. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEES-92-061
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 10/12/92

EVALUATION FOR TSA 03-92-06-12 FIRE WATER PUMP TRIP UPON LOOP DURING 4160 VOLT BUS 3A DE-ENERGIZATION

Summary:

For the Unit 3 refueling outage, this evaluation was developed in support of Temporary System Alteration (TSA) 03-92-06-12, which provided a trip circuit scheme for the electric-driven Fire Water Pump (FWP) during the 3A 4kV bus outage. The FWP was powered from the 480 volt Load Center (LC) 3C. The FWP was designed to trip upon the loss of voltage; however, the trip circuit would be disabled when the 3A 4kV load sequencer was removed from service as part of the 3A 4kV bus outage, and therefore allow the FWP to be auto-connected to EDG 3B in the first load block. The TSA required that wires of sufficient length be installed within LC 3C between spare contacts of two undervoltage relays, and FWP breaker control circuit. This would preclude the FWP from being automatically loaded onto Emergency Diesel Generator 3B upon initiation of a loss of offsite power.

Safety Evaluation:

The temporary use of an alternate relay provided undervoltage protection to trip the fire water pump breaker open in the event of an undervoltage condition restored compliance with the design basis for the fire water pump, while the 3A 4kV bus was de-energized for maintenance. The installation of a temporary jumper did not adversely interact with the 3B EDG or any equipment important to safety. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECS-92-063
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 10/30/92

THE INSTALLATION OF COMMUNICATION ANTENNAS (TP-907)

Summary:

This evaluation addressed the acceptability of installing two antennas; one fiberglass whip antenna on the control room roof and one loop antenna on the Unit 4 EDG Building. As a result of Hurricane Andrew, offsite communications were interrupted due to loss of all communication paths from the site because of equipment damage. A comprehensive wireless system was being considered for installation in order to preclude communication losses in the future. Under Test Procedure TP-907, various communications tests were performed to assess the feasibility and performance of various antenna/radio systems. The subject antennas were installed on a temporary basis in order to accumulate test data pertaining to the acceptability for proposed antenna locations.

Safety Evaluation:

This safety evaluation addressed the mounting of the antennas and their potential interaction with safety related structures, systems and equipment and concluded that the antennas can be installed provided that all of the requirements stipulated within this evaluation were followed. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-92-066
REVISION 0

UNIT : 3
APPROVAL DATE : 11/05/92

FREEZE SEAL SAFETY EVALUATION FOR REPAIR OF CV-3-244

Summary:

This safety evaluation addressed the use of a freeze seal in order to repair valve CV-3-244 at the discharge of the Chemical and Volume Control System (CVCS) demineralizers. In order to perform corrective maintenance, a freeze seal was utilized to provide isolation from the letdown bypass path around the CVCS demineralizers. The purpose of the freeze seal was to allow for the continued use of letdown. This maintenance was performed during Modes 5 and/or 6.

Safety Evaluation:

During the maintenance activity, equipment important to safety required for accident mitigation remained available to perform its required safety functions. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECS-92-070
REVISION 1

UNIT : 4
APPROVAL DATE : 11/24/92

REPLACEMENT OF CRDM 4A COOLER FAN MOTOR AT POWER OPERATION

Summary:

This evaluation addressed the acceptability of replacing the 4A CRDM cooler fan motor while Unit 4 was in power operation (Mode 1). The evaluation addressed the use of the Polar Crane in Mode 1 operation, including identification of safe load paths, consequences of load drops on safety related equipment, and seismic considerations. It further addressed the use of scaffolding and radiation shielding including adverse seismic interactions with safety related equipment, the effect of high energy line break jet impingements, and other potentials for adverse interactions. Finally, it addressed the effects of the removal of the fan plenum and motor on the structural integrity of the CRDM cooler ductwork and associated CCW lines.

Revision 1 of this evaluation provided additional clarification of the response to concerns related to the potential for sump screen blockage by debris.

Safety Evaluation:

This evaluation concluded that this activity would have no adverse impact on the plant operations, and would not compromise the safety and licensing bases for Turkey Point Unit 4. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SECS-92-071
REVISION 0

UNIT : 3
APPROVAL DATE : 11/20/92

SAFETY EVALUATION FOR ALLOWING A MAN-BASKET TO REMAIN WITHIN CONTAINMENT DURING ALL MODES OF OPERATION

Summary:

This evaluation addressed the acceptability of leaving a man-basket within the Unit 3 Containment Structure during all modes of operation. The man-basket in combination with the Polar Crane was utilized during the Unit 3 Cycle 13 refueling outage for maintenance activities and for valve manipulations in preparation for the integrated leak rate test (ILRT). In order to remove the man-basket, the containment equipment hatch would be required to be opened. Due to schedular considerations Nuclear Engineering investigated the acceptability of allowing the basket to remain within containment during all modes of operation. This safety evaluation concluded that the man-basket can remain within the containment structure during all modes of operation provided that all of the requirements stipulated within this evaluation were followed.

Safety Evaluation:

The storage of a steel man-basket secured to structural steel on the 58 foot elevation within containment will not interact with any equipment that performs a safety function. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-92-072
REVISION 0

UNIT	:	3
TURNOVER DATE	:	11/27/92

SAFETY EVALUATION
FOR LT-3-494 VENT PATH MODIFICATION

Summary:

This safety evaluation was written to support the plant changes implemented under PC/M 92-176, whose implementation was completed and turned over to the plant by November 27, 1992. This PC/M was developed to address the modification of a steam generator level transmitter (LT-3-494) vent path by the removal of valve 3-20-802, which was replaced with a pipe cap. Valve 3-20-802 had been identified as requiring replacement during the ongoing refueling outage. During the analysis of this replacement valve, it was noted that the stress in the line containing the two series vent valves did not meet FSAR allowable stresses. To correct this condition, removal of the top most vent valve, 3-20-802, was required. This valve was replaced with a pipe cap, which served to to provide the same isolation function as the original valve.

Safety Evaluation:

Valve 3-20-803 was the primary pressure boundary for the steam generator level transmitter LT-3-494 and the installed pipe cap provided the same backup pressure isolation as the original valve 3-20-802. The changes identified and evaluated in this safety evaluation did not have any adverse impact on plant safety or plant operations. The hardware changes did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the changes identified in this PC/M and safety evaluation.

SAFETY EVALUATION JPN-PTN-SENS-93-007
REVISION 0

UNIT : 4
APPROVAL DATE : 03/30/93

TEMPORARY REMOVAL OF STEAM GENERATOR 4C THRUST BEAM

Summary:

This safety evaluation established requirements for the temporary removal and reinstallation of structural components to accommodate the Reactor Coolant Pump (RCP) motor replacement during refueling outages. To provide adequate clearance for rigging motors through the Unit 4 Containment equipment hatch and facilitate staging for the refueling outage, the Steam Generator 4C thrust beam, floor steel, handrail, grating and pipe supports for the 2-inch containment primary water service connections and 2-inch containment service air piping above the equipment hatch must be temporarily removed. Following the outage all components were replaced.

Safety Evaluation:

No permanent change in the plant configuration was involved. The structural items removed were reinstalled to the same configuration and to the same design requirements as the original installation. The effects on existing systems, structures, and components due to the temporary removal of these structural items were evaluated with respect to plant operational modes. The temporary changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The temporary plant modifications, identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the temporary changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-009
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 03/16/93

MACHINING OF MOTOR OPERATED VALVE STEMS FOR INSTALLATION OF STRAIN GAUGES - SPECIFICATION SPEC-M-009

Summary:

This evaluation provided the basis for the acceptability of using SPEC-M-009 in the maintenance process. FPL Specification, SPEC-M-009, "Machining of Motor Operated Valve Stems for Stain Gauge Installation" provided engineering guidance and details sufficient to allow field machining of threaded valve stem sections for installation of Teledyne miniature strain gauges. These strain gauges were provided in support of NRC Generic Letter, 89-10 concerning MOV actuator load monitoring. By utilizing the specification in lieu of an engineering package greater flexibility in the implementation process resulted. This specification allowed all or part of the identified valve scope to be implemented, and additional valve scope could be added in the future by specification and corresponding calculation revisions, if desired.

Safety Evaluation:

This evaluation concluded that the method of implementation and limitations imposed by SPEC-M-009 are consistent with all associated technical and licensing requirements. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-010
REVISIONS 0 & 1

UNIT	:		4
APPROVAL DATES	:	Rev.0	03/25/93
		Rev.1	04/08/93

INTAKE COOLING WATER VALVE REPLACEMENTS AND B HEADER CRAWL THROUGH INSPECTION

Summary:

The purpose of this safety evaluation was to demonstrate that there was no adverse effect on plant safety or operations associated with the replacement of eight Intake Cooling Water (ICW) valves and the crawl through inspection/repair of ICW piping. Eight Intake Cooling Water (ICW) isolation valves were replaced due to excessive leakage during the 1993 Unit 4 refueling outage. A crawl through inspection and repair of the Unit 4 B ICW header and the C ICW pump discharge piping was also performed during the same outage. Some of the valve replacements required one of the ICW headers to be removed from service. To ensure that Residual Heat Removal (RHR) and spent fuel pool cooling requirements continued to be met, ICW operations were controlled in accordance with the applicable Technical Specifications and system operating procedures.

Revision 1 of this safety evaluation added the replacement of valve 4-50-340 to the scope of this safety evaluation. The additional scope was warranted based on the results of leak testing performed after the issuance of Revision 0.

Safety Evaluation:

The ICW valve replacement and crawl through activities did not adversely affect the operation of equipment important to safety necessary to support any Mode of operations. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-011
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 05/11/93

OMS SETPOINT DURING RCP OPERATION

Summary:

The purpose of this safety evaluation was to assess the ability of the Overpressure Mitigating System (OMS) to provide protection against the system design basis overpressurization events during RCP operation. Recent industry findings on the methodology of determining Overpressure Mitigation System (OMS) setpoints prompted a review of the setpoints at Turkey Point. During the review, it was determined that the calculations for determining the setpoints did not consider pressure differences between the reactor vessel and the pressure transmitters caused by RCP operation and elevation differences.

To ensure that in all cases no overpressure transient could occur, restrictions were imposed to either decrease the PORV stroke times or to limit RCP operations during cold, water solid operations. These actions, in conjunction with ASME Code Case N-514 (which allows primary pressure to reach up to 110% of the pressure/temperature limits during cold overpressure events), assured that the OMS was operable and capable of protecting the reactor vessel from damage from all postulated cold overpressure transients.

Safety Evaluation:

Reactor coolant pumps do not contribute to any accident mitigation analyses in Mode 5. A shorter PORV open stroke time does not adversely impact any previously postulated accident in Mode 5, and serves to mitigate those accidents addressed within the safety evaluation. The actions and changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions and plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions and changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEEP-93-015
REVISION 0

UNIT	:	4
APPROVAL DATE	:	05/19/93

SAFETY EVALUATION FOR ACCEPTABLE UPPER AND LOWER TIME DELAY LIMITS FOR ECC 4A AND ECF 4A AGASTAT LOAD SEQUENCING RELAYS

Summary:

This safety evaluation provided acceptance criteria for ECC 4A and ECF 4A load block sequencer timing. During Engineered Safeguards Integrated Testing the two Agastat relays associated with sequencing the Emergency Containment Cooler (ECC) 4A and Emergency Containment Filter (ECF) 4A failed to meet the existing test acceptance criteria. The as-left setting for the ECF relay was 37.5 seconds. This safety evaluation established a basis for the as-left settings of the Agastat relays and provided acceptance criteria for future testing commensurate with equipment accuracies.

Safety Evaluation:

This safety evaluation demonstrated that the ECF and ECC fans will start and reach operating speeds within the time limits prescribed in the most limiting plant accident analyses. In addition, acceptance criteria for future testing was consistent with the most limiting design basis accident analyses. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-017

REVISION 0

UNIT : 4
APPROVAL DATE : 05/06/93

EVALUATION FOR LOOSE OBJECTS IN THE SECONDARY SIDE OF STEAM GENERATOR C AT TURKEY POINT UNIT 4

Summary:

This evaluation addressed the potential safety significance of operating Turkey Point Unit 4 Steam Generator 'C', with loose objects (screws) present in the secondary side. These screws were described as three (3) round head screws, which attached an inspection camera light bracket to its camera housing. The bracket was located and retrieved from the tube lane (blowdown lane), however, thorough remote visual inspection of the tube lane did not reveal the screws. These screws were presumed to be in the tube lane and most probably below the blowdown pipe where visual contact could not be made. Their location was presumed to be between two (2) flow restrictor baffles located at Columns 72 and 79.

Safety Evaluation:

Analysis showed that any potential tube wear from the screws would not occur beyond a depth equivalent to the current Technical Specification plugging limit of 40% wall loss. The screws were not expected to exit the steam generator and enter the Main Steam System and, therefore, will not impact any accident analysis that considers the Main Steam System. It was also determined that isolation of the Steam Generator Blowdown and Sampling System would not be adversely impacted by the screws. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-018
REVISION 0

UNIT	:	4
APPROVAL DATE	:	05/01/93

SAFETY EVALUATION FOR STEAM GENERATOR C SECONDARY SIDE FOREIGN OBJECTS

Summary:

This evaluation addressed the effects of a foreign object identified on the tube sheet surface of the Unit 4 'C' Steam Generator. The object was a piece of wire approximately 3" long and 1/8" in diameter, and had been determined to be irretrievable. Previous Eddy Current Test (ECT) data confirms that this object has remained lodged in the same location since the previous refueling outage. An ECT performed during this outage further shows that no damage has occurred to the tubes adjacent to the object due to it's presence. The purpose of this safety evaluation was to assess the acceptability of resuming Unit 4 operation with the foreign object remaining lodged in the C Steam Generator.

Safety Evaluation:

This safety evaluation determined that the object had been fixed in its present location for at least one full operating cycle and that no damage to the adjacent tubes had resulted. Based on this documented experience, future movement of the object was not expected. The generator would not be damaged by the foreign object during future operation. However, continued monitoring of the object would be performed to ensure that the conclusions of this safety evaluation remained valid during subsequent fuel cycles. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-019
REVISION 0

UNIT : 4
APPROVAL DATE : 05/01/93

SAFETY EVALUATION FOR STEAM GENERATOR A SECONDARY SIDE FOREIGN OBJECTS

Summary:

This evaluation addressed the potential safety impact of continued operation of the Turkey Point Unit 4 plant with a potentially mobile foreign object remaining in the secondary side of Steam Generator A. During a routine foreign object search and retrieval operation, a total of 4 foreign objects were detected. Three of the four identified objects were retrieved, and only one object remained. This object was described as a flat washer with a nut integral to the washer. In the worst case, foreign objects in the steam generator secondary side could cause significant tube wear, tube wear with primary to secondary leakage and possibly a potential tube rupture event.

Safety Evaluation:

This evaluation demonstrated that operation of the steam generators with the identified foreign objects remaining in the steam generators would not have an adverse effect on the pressure boundary integrity of the steam generators. The actions or changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or plant changes (procedures and/or hardware), identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified in this safety evaluation.

SECTION 3

RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 91-108

UNIT	:	3
TURN OVER DATE	:	12/18/92

TURKEY POINT UNIT 3 CYCLE 13
RELOAD SAFETY AND LICENSING CHECKLIST

Summary:

This engineering package provided the reload core design of the Turkey Point Unit 3 Cycle 13. This engineering design also extended the service life of the Hafnium Vessel Flux Depressor (HVFD) clusters to the end of Cycle 13. The primary design change to the core for Cycle 13 was the replacement of 57 irradiated assemblies with 56 fresh Region 15 fuel assemblies and 1 irradiated assembly reinserted from Cycle 8. The fuel was arranged in a low leakage pattern with no significant differences between the Cycle 13 loading pattern and the Cycle 12 design. Cycle 13 also marked the elimination of secondary neutron sources in Turkey Point Unit 3. Region 15 used the same Debris Resistant Fuel Assemblies (DRFA) as Cycle 12, except for several minor design changes.

Safety Evaluation:

The design of Turkey Point Unit 3 Cycle 13 was evaluated by Westinghouse. The Cycle 13 reload core design, including the reconstituted fuel assemblies, met all applicable design criteria and all pertinent licensing basis. The minor design modifications to the fuel assembly and core components (WABA) did not affect the applicable design criteria for these components. These changes had no impact on fuel rod performance, dimensional stability or core operating limits. The extension of the residence time of the HVFD rods likewise did not impact their performance or exceed their design criteria. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 92-045

UNIT	:	4
TURN OVER DATE	:	05/21/93

TURKEY POINT UNIT 4 CYCLE 14
RELOAD SAFETY AND LICENSING CHECKLIST

Summary:

This engineering package provided the reload core design of the Turkey Point Unit 4 Cycle 14. The primary design change to the core for Cycle 14 was the replacement of 52 irradiated assemblies with 52 fresh Region 16 fuel assemblies. These fresh assemblies were Debris Resistant Fuel Assemblies (DRFA) and all contain a 6-inch axial blanket of .71 w/o U^{235} (natural uranium) at both the top and bottom of the fuel stack. This was the first use of axial blankets in Unit 4. The fuel was arranged in a low leakage pattern with no significant differences between the Cycle 14 loading pattern and the Cycle 13 design. Region 16 used the same Debris Resistant Fuel Assemblies (DRFA) as Cycle 13, except for the following design changes which included : 1) the use of axial blankets; 2) implementation of an anti-snag outer grid strap in the top and bottom inconel grids; and 3) a change to the Wet Annular Burnable Absorber (WABA) rodlet spacer length. The spacer within the WABA rodlet assembly was lengthened to shift the WABA absorber stack upward to align the center of the absorber stack with the fuel midplate.

Safety Evaluation:

The design of Turkey Point Unit 4 Cycle 14 was evaluated by Westinghouse. The Cycle 14 reload core design met all applicable design criteria and all pertinent licensing bases. The minor design modifications to the fuel assembly and core components (WABA) did not affect the applicable design criteria for these components. These changes had no impact on fuel rod performance, dimensional stability or core operating limits. The extension of the residence time of the HVFD rods likewise did not impact their performance or exceed their design criteria. The modification in this Engineering Package did not have any adverse effect on plant safety or plant operations. This modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

SECTION 4

ANNUAL REPORT OF POWER OPERATED RELIEF VALVE (PORV) ACTUATIONS

ANNUAL REPORT OF SAFETY AND RELIEF VALVE CHALLENGES

By letter dated June 18, 1980 (L-80-186) Florida Power and Light stated the intent to comply with the requirements of item II.K.3.3 of enclosure 3 to the commissioner's letter of May 7, 1980 (Five Additional TMI-2 Related Requirements to Operating Reactors).

The following is a list of power operated relief valve (PORV) challenges for Turkey Point Units 3 and 4 from July 1, 1992 to June 1, 1993.

Unit 3

November 27, 1992	With the unit in Mode 6, PCV-3-456 actuated twice due to high RCS pressure which occurred as a result of starting the 3C Reactor Coolant Pump.
January 16, 1993	A delay in stopping the charging while the pressurizer was being filled resulted in an RCS pressure increase which caused an actuation of PORV PCV-3-456.

UNIT 4

September 19, 1992	With Unit 4 in Mode 3, PORV PCV-4-456 opened during testing of valves MOV-4-750 and MOV-4-751.
October 5, 1992	With Unit 4 in Mode 5, PORV PCV-4-455C opened during a surveillance of the Overpressure Mitigating System, resulting in slight depressurization of the Reactor Coolant System.

SECTION 5

STEAM GENERATOR TUBE INSPECTIONS FOR TURKEY POINT

Eddy Current Summary of Results					
Plant: Turkey Point 3					
Examination Dates: 10/10/92 Through 10/18/92					
Steam Generator Number	Total Tubes Inspected	Total Ind. ≥ 20% to 39%	Total Ind. ≥ 40% to 100%	Total Tubes Plugged as Preventive Maintenance	Total Tubes Plugged
3E210A	3199	72	1	NONE	1
3E210B	3200	95	NONE	1	1
3E210C	3188	73	3	2	5

Location of Indications

Steam Generator	AVB Bars	Drilled Support 1 through 6		Top of Tube Sheet to 1 Drilled Support	
		Cold Leg	Hot Leg	Cold Leg	Hot Leg
3E210A	37	13	14	5	4
3E210B	44	22	17	NONE	12
3E210C	59	6	6	3	2

Certification of Record

We certify that the statements in this record are correct and the tubes inspected were tested in accordance with the requirements of Section XI of the ASME Code.

FLORIDA POWER and LIGHT COMPANY
Organization

Date: 12/7/92 Prepared By: Wolfgang K Heise
S/G Eddy Current Coordinator

Date: 12/8/92 Reviewed By: [signature]
Inspections Supervisor

Steam Generator Tubes Plugged								
Plant: Turkey Point 3								
Steam Generator 3E210A			Steam Generator 3E210B			Steam Generator 3E210C		
Row	Column	Remarks	Row	Column	Remarks	Row	Column	Remarks
21	32	44%	42	45	PTP	33	39	PTP
						35	41	PTP
						2	55	60%
						20	66	41%
						2	70	45%

PTP - Preventive tube plug based on memos attached to CNR # 92-3-031 and 032.

The following cumulative listings are attached for each steam generator.

- Cumulative Distribution Summary
- Pluggable Indications
- Indications 20 - 39%

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/92

COMPONENT : S/G A

Page : 1 of 1
Date : 11/16/92
Time : 11:30 AM

Examination Dates : 10/10/92 thru 10/18/92

Total Number of Tubes Inspected .....: 3199

Total Indications
Between 20% and 39% ....: 72
Greater than or equal to 40% ...: 1

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 1

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	4	TSC -.5 to 01C -2.1 :	5
01H -2.0 to 06H +2.0 :	14	01C -2.0 to 06C +2.0 :	13
06H +2.1 to AV1 -3.1 :	21	06C +2.1 to AV4 -3.1 :	3
AV1 -3.0 to AV4 -3.0 :	13		

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G A
DESCRIPTION : PLUGGABLE INDICATIONS

Date : 11/13/92
Time : 9:00 AM

			Extent				09/92						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
21	32	C	TEH	TEH PS	AC012-03	A-720-M/ULC	06H 2.3	1.3	135	1	44						

Number of RECORDS Selected from Current Outage : 1
Number of TUBES Selected from Current Outage : 1

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G A
DESCRIPTION : 20% TO 39%

Date : 11/13/92
Time : 9:00 AM

Row	Col	Leg	Extent Req	Tst/Note	Reel	Probe	Location	09/92 Volts	Deg	Ch	%	Diff	N/A Location	Volts	Deg	Ch	%
2	4	C	06C	06C	AC001-01	A-720-M/ULC	05C 44.1	.5	149	1	30						
9	4	C	TEH	TEH PS	AC004-01	A-720-M/ULC	06H 2.4	.6	151	1	30						
13	5	C	TEH	TEH	AC006-02	A-720-M/ULC	TSH 3.6	1.1	157	1	26						
		C	TEH	TEH	AH030-07	MRPC720-3C/7	TSH 3.9	1.4	131	1	32						
9	6	C	TEH	TEH	AC004-01	A-720-M/ULC	06H 2.5	.4	150	1	31						
14	7	C	TEH	TEH PC	AC006-02	A-720-M/ULC	TSH 2.7	.2	152	1	31						
9	11	C	TEH	TEH	AC004-01	A-720-M/ULC	06H 2.5	.7	149	1	32						
9	12	C	TEH	TEH	AC005-01	A-720-M/ULC	06H 2.4	1.2	160	1	22						
9	13	C	TEH	TEH PS	AC005-01	A-720-M/ULC	06H 2.4	.6	148	1	33						
19	13	C	TEH	TEH PC	AC007-02	A-720-M/ULC	02C 21.5	.7	140	1	39						
9	14	C	TEH	TEH	AC005-01	A-720-M/ULC	06H 2.3	.8	152	1	30						
10	16	C	TEH	TEH	AC005-01	A-720-M/ULC	02C 23.9	.9	146	1	35						
9	18	C	TEH	TEH	AC005-01	A-720-M/ULC	06H 2.4	.8	158	1	24						
9	20	C	TEH	TEH	AC005-01	A-720-M/ULC	06H 2.3	1.1	151	1	30						
9	21	C	TEH	TEH	AC005-01	A-720-M/ULC	06H 2.2	1.0	152	1	30						
9	23	C	TEH	TEH SS	AC006-02	A-720-M/ULC	06H 2.3	.8	148	1	34						
22	30	C	TEH	TEH PC	AC012-03	A-720-M/ULC	04H 28.8	.6	162	1	21						
23	31	C	TEH	TEH SS	AC012-03	A-720-M/ULC	01H 12.9	.8	160	1	23						
2	32	C	06C	06C PS	AC001-01	A-720-M/ULC	05C 43.1	.5	158	1	22						
17	32	H	TEC	TEC PC	AH028-06	A-720-M/ULC	01C 10.6	.5	154	000	29						
17	34	H	TEC	TEC PS	AH028-06	A-720-M/ULC	01H 24.4	.4	151	1	31						
18	37	H	TEC	TEC	AH029-07	A-720-M/ULC	01C 3.1	.7	158	1	25						
19	37	H	TEC	TEC	AH029-07	A-720-M/ULC	01C 3.1	.9	160	1	23						
21	38	C	TEH	AV3	AH032-08	MRPC680-5FH	AV2 12.5	2.2	147	1	25						
29	40	C	TEH	TEH PS	AC014-04	A-720-M/ULC	02H 18.8	.6	160	1	23						
33	41	C	TEH	TEH PC	AC018-05	A-720-M/ULC	AV1 .0	.8		P 2	25						
		C	TEH	TEH PC	AC018-05	A-720-M/ULC	AV3 .0	.5		P 2	23						
31	44	C	TEH	TEH PS	AC019-05	A-720-M/ULC	AV3 .0	.5		P 2	25						
17	45	H	TEC	TEC	AH029-07	A-720-M/ULC	02H 3.9	1.0	155	1	28						
38	45	C	TEH	TEH PS	AC019-05	A-720-M/ULC	AV2 .0	.6		P 2	26						
		C	TEH	TEH PS	AC019-05	A-720-M/ULC	AV3 .0	.5		P 2	25						
41	46	H	TEC	TEC PS	AH033-08	A-720-M/ULC	06H 5.4	.6	150	1	32						
24	47	H	TEC	TEC PS	AH017-03	A-720-M/ULC	AV1 .0	1.4		P 2	29						
37	47	C	TEH	TEH PS	AC019-05	A-720-M/ULC	AV3 .0	.4		P 2	24						
22	51	H	TEC	TEC PS	AH027-06	A-720-M/ULC	06C 3.5	.4	144	1	37						
22	52	H	TEC	TEC	AH018-03	A-720-M/ULC	04H 40.9	.9	141	1	39						
		H	TEC	TEC	AH030-07	MRPC720-3C/7	04H 42.4	.9	116	1	39						
30	52	H	TEC	TEC PS	AH018-03	A-720-M/ULC	AV3 .0	.4		P 2	23						
2	54	C	06C	06C PS	AC002-01	A-720-M/ULC	01C 3.3	.4	163	1	21						
15	55	H	TEC	TEC PS	AH025-05	A-720-M/ULC	TSC 8.3	1.2	149	1	34						
30	58	H	TEC	TEC	AH019-04	A-720-M/ULC	AV1 .0	1.5		P 2	30						
		H	TEC	TEC	AH019-04	A-720-M/ULC	AV2 .0	1.0		P 2	27						
		H	TEC	TEC	AH019-04	A-720-M/ULC	AV3 .0	1.5		P 2	30						

COMPONENT : S/G A
DESCRIPTION : 20% TO 39%

			Extent					09/92					N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
		H	TEC	TEC	AH019-04	A-720-M/ULC	AV4 .0	.7		P 2	25						
28	59	H	TEC	TEC	AH019-04	A-720-M/ULC	AV2 .0	.7		P 2	24						
		H	TEC	TEC	AH019-04	A-720-M/ULC	AV3 .0	.6		P 2	24						
29	59	H	TEC	TEC	AH019-04	A-720-M/ULC	05H 25.7	1.2	160	1	23						
		H	TEC	TEC	AH019-04	A-720-M/ULC	05C 36.4	1.3	162	1	21						
41	59	H	TEC	TEC PS	AH034-08	A-720-M/ULC	06H 4.6	.5	148	1	34						
38	65	C	TEH	TEH SC	AC022-05	A-720-M/ULC	AV2 .0	.7		P 3	23						
9	67	H	TEC	TEC PS	AH009-02	A-720-SF/RM	TSC 46.0	.9	149	1	32						
20	68	H	TEC	TEC PS	AH027-06	A-720-M/ULC	04H 42.2	.3	159	1	24						
19	69	H	TEC	TEC PS	AH027-06	A-720-M/ULC	03C 41.0	.3	149	1	33						
27	70	H	TEC	TEC	AH021-04	A-720-M/ULC	01H 45.8	1.0	159	1	24						
30	71	H	TEC	TEC PC	AH021-04	A-720-M/ULC	04C 2.8	.3	145	1	37						
1	73	C	06C	06C	AC003-01	A-720-M/ULC	BAC 13.6	.5	147	1	34						
37	73	H	TEC	TEC	AH023-05	A-720-M/ULC	04H 36.8	.5	148	1	34						
4	74	H	TEC	TEC	AH011-01	A-720-M/ULC	03H 9.1	1.6	145	1	35						
32	75	H	TEC	TEC	AH023-05	A-720-M/ULC	AV3 2.1	.8	143	P 2	25						
9	76	H	TEC	TEC SS	AH011-01	A-720-M/ULC	03C 43.1	.4	143	1	38						
		H	TEC	TEC	AH011-01	A-720-M/ULC	02C 16.7	1.0	144	1	37						
1	81	C	06C	06C	AC003-01	A-720-M/ULC	BAC 16.0	.5	155	1	27						
22	81	H	TEC	TEC	AH022-05	A-720-M/ULC	05H 40.5	.3	138	P 1	24						
9	82	H	TEC	TEC	AH012-02	A-720-M/ULC	06H 2.9	.9	146	1	35						
24	82	H	TEC	TEC SC	AH022-05	A-720-M/ULC	06C 4.1	1.1	158	1	25						
9	83	H	TEC	TEC SS	AH012-02	A-720-M/ULC	06H 2.6	1.1	145	1	36						
1	84	C	06C	06C PS	AC003-01	A-720-M/ULC	BAC 16.6	1.2	157	1	25						
19	84	H	TEC	TEC	AH015-03	A-720-M/ULC	TSH .9	.4	159	1	22						
9	85	H	TEC	TEC SS	AH012-02	A-720-M/ULC	06H 2.4	.9	144	1	37						
9	86	H	TEC	TEC	AH012-02	A-720-M/ULC	06H 2.3	1.7	162	1	20						
9	90	H	TEC	TEC	AH012-02	A-720-M/ULC	06H 2.4	1.2	159	1	23						
12	90	H	TEC	TEC PS	AH016-03	A-720-M/ULC	04H -.5	.1	124	P 1	32						

Number of RECORDS Selected from Current Outage : 72
Number of TUBES Selected from Current Outage : 62

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/92

COMPONENT : S/G B

Page : 1 of 1
Date : 11/16/92
Time : 11:30 AM

Examination Dates : 10/10/92 thru 10/18/92

Total Number of Tubes Inspected .....: 3200

Total Indications
Between 20% and 39% ....: 95
Greater than or equal to 40% ...: 0

Total Tubes Plugged as Preventive Maint : 1
Total Tubes Plugged ....: 1

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	12	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	17	01C -2.0 to 06C +2.0 :	22
06H +2.1 to AV1 -3.1 :	13	06C +2.1 to AV4 -3.1 :	10
AV1 -3.0 to AV4 -3.0 :	21		

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G B
DESCRIPTION : PLUGGABLE INDICATIONS

Date : 11/13/92
Time : 9:00 AM

			Extent				09/92						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
42	45	C	TEH	TEH *N	BC025-07	A-720-M/ULC	AV2 .0	2.2		35%	PTP						

Number of RECORDS Selected from Current Outage : 1
Number of TUBES Selected from Current Outage : 1

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G B
DESCRIPTION : 20% TO 39%

Date : 11/13/92
Time : 9:00 AM

			Extent				09/92						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
11	2	C	TEH	TEH PS	BC008-03	A-720-SF/RM	02C 37.0	.5	147	1	30						
10	5	C	TEH	TEH	BC006-02	A-720-M/ULC	02C 31.8	.8	154	1	22						
21	6	C	TEH	TEH PC	BC011-03	A-720-M/ULC	05H 34.3	.3	135	1	38						
10	7	C	TEH	TEH	BC006-02	A-720-M/ULC	02C 31.7	.6	155	1	21						
5	8	C	TEH	TEH	BC006-02	A-720-M/ULC	BAH 15.9	.7	156	1	20						
		C	TEH	TEH	BC006-02	A-720-M/ULC	04H 15.0	1.8	154	1	22						
8	8	C	TEH	TEH	BC006-02	A-720-M/ULC	01H .7	.6	123	P 1	30						
11	9	C	TEH	TEH PC	BC009-03	A-720-SF/RM	02C 36.6	.4	141	1	36						
23	9	C	TEH	TEH	BC011-03	A-720-M/ULC	06H 4.0	1.4	143	1	32						
		C	TEH	TEH	BC011-03	A-720-M/ULC	05C 31.2	.7	154	1	22						
		C	TEH	TEH	BC011-03	A-720-M/ULC	05C 18.5	.3	140	1	36						
11	10	C	TEH	TEH PS	BC009-03	A-720-SF/RM	02C 36.7	.6	141	1	36						
23	10	C	TEH	TEH	BC011-03	A-720-M/ULC	06H 4.5	1.3	144	1	33						
10	11	C	TEH	TEH PS	BC006-02	A-720-M/ULC	02C 31.9	.3	145	1	30						
11	11	C	TEH	TEH SS	BC009-03	A-720-SF/RM	02C 36.7	.7	157	1	20						
18	12	C	TEH	TEH PC	BC009-03	A-720-SF/RM	05H 19.8	.5	141	1	36						
19	12	C	TEH	TEH PS	BC009-03	A-720-SF/RM	TSH .7	.6	137	1	39						
29	12	C	TEH	TEH PS	BC011-03	A-720-M/ULC	AV1 .0	.4		P 2	23						
20	13	C	TEH	TEH PC	BC009-03	A-720-SF/RM	04H 15.4	.3	148	1	29						
6	14	C	TEH	TEH SS	BC006-02	A-720-M/ULC	02H 32.2	.9	144	1	31						
33	15	C	TEH	TEH PS	BC016-05	A-720-M/ULC	AV2 .0	.5		P 2	21						
36	19	C	TEH	TEH PS	BC016-05	A-720-M/ULC	AV2 .0	.5		P 2	21						
26	20	C	TEH	TEH PS	BC012-04	A-720-M/ULC	AV4 .0	.5		P 2	23						
37	20	C	TEH	TEH PS	BC016-05	A-720-M/ULC	TSH .6	1.5	145	1	30						
33	21	C	TEH	TEH SS	BC016-05	A-720-M/ULC	06H 4.8	1.0	154	1	22						
33	23	C	TEH	TEH SS	BC016-05	A-720-M/ULC	06H 5.0	.7	136	1	38						
40	25	C	TEH	TEH	BC017-05	A-720-M/ULC	05C 39.3	.5	145	1	30						
33	26	C	TEH	TEH PC	BC017-05	A-720-M/ULC	06H 5.1	.6	141	1	33						
40	26	C	TEH	TEH PS	BC017-05	A-720-M/ULC	AV3 .0	.3		P 2	21						
40	27	C	TEH	TEH PS	BC017-05	A-720-M/ULC	AV2 .0	.5		P 2	23						
28	28	C	TEH	TEH PS	BC013-04	A-720-M/ULC	03C 34.2	1.0	135	1	38						
33	29	C	TEH	TEH PS	BC017-05	A-720-M/ULC	06H 5.0	.5	139	1	35						
39	30	C	TEH	TEH SS	BC017-05	A-720-M/ULC	02C 45.7	.5	154	1	21						
11	31	H	TEC	TEC PS	BH005-01	A-720-M/ULC	02C 37.2	.3	151	1	26						
34	31	C	TEH	TEH PS	BC017-05	A-720-M/ULC	AV2 .0	.4		P 2	22						
6	32	H	TEC	TEC	BH025-07	A-720-M/ULC	TSH 39.0	.8	146	1	32						
5	34	H	TEC	TEC	BH025-07	A-720-M/ULC	TSH 31.6	.6	151	1	27						
6	34	H	TEC	TEC PS	BH025-07	A-720-M/ULC	TSH 38.4	.4	137	1	39						
32	34	C	TEH	TEH SS	BC017-05	A-720-M/ULC	AV3 .0	.8		P 3	22						
		C	TEH	TEH SS	BC017-05	A-720-M/ULC	AV4 .0	.6		P 3	20						
5	35	H	TEC	TEC	BH025-07	A-720-M/ULC	TSH 33.5	.5	140	1	36						
6	36	C	TEH	TEH PC	BC027-08	A-720-M/ULC	TSH 37.5	.7	136	1	38						
44	36	C	TEH	TEH SS	BC025-07	A-720-M/ULC	AV1 .0	1.0		P 3	21						

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G B
DESCRIPTION : 20% TO 39%

Date : 11/13/92
Time : 9:00 AM

Row	Col	Leg	Extent Req	Extent Tst/Note	Reel	Probe	09/92 Location	09/92 Volts	09/92 Deg	09/92 Ch	09/92 %	Diff	N/A Location	N/A Volts	N/A Deg	N/A Ch	N/A %
14	37	H	TEC	TEC PS	BH005-01	A-720-M/ULC	02C 46.5	.2	151	1	26						
42	37	C	TEH	TEH PS	BC025-07	A-720-M/ULC	TSH .8	1.0	138	1	37						
44	37	C	TEH	TEH PS	BC025-07	A-720-M/ULC	AV4 .0	.4		P 2	23						
34	38	C	TEH	TEH SS	BC019-06	A-720-M/ULC	AV2 .0	2.2		P 3	31						
		C	TEH	TEH SS	BC019-06	A-720-M/ULC	AV3 .0	1.8		P 3	30						
42	38	C	TEH	TEH	BC025-07	A-720-M/ULC	TSH 1.4	1.1	150	1	25						
		C	TEH	TEH	BC025-07	A-720-M/ULC	TSH 3.2	.5	149	1	26						
11	39	H	TEC	TEC PS	BH006-02	A-720-M/ULC	01H 12.0	.5	144	1	32						
39	39	C	TEH	TEH	BC019-06	A-720-M/ULC	05H .8	1.5	111	P 1	39						
44	40	C	TEH	TEH PS	BC025-07	A-720-M/ULC	AV1 .0	.5		P 2	24						
14	42	H	TEC	TEC PS	BH006-02	A-720-M/ULC	03H 21.1	.5	156	1	21						
44	42	C	TEH	TEH PS	BC025-07	A-720-M/ULC	AV1 .0	.6		P 3	20						
6	44	C	TEH	TSH PS	BC027-08	A-720-M/ULC	TSH 39.3	1.6	142	1	32						
19	44	H	TEC	TEC SS	BH006-02	A-720-M/ULC	02H 32.0	.8	139	1	36						
42	45	C	TEH	TEH	BC025-07	A-720-M/ULC	AV2 .0	2.2		P 2	35						
		C	TEH	TEH	BC025-07	A-720-M/ULC	AV3 .0	.9		P 2	28						
34	46	C	TEH	TEH PC	BC020-06	A-720-M/ULC	AV2 .0	.9		P 2	28						
		C	TEH	TEH PC	BC020-06	A-720-M/ULC	AV3 .0	1.5		P 2	31						
35	48	C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV2 .0	.5		P 2	26						
		C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV3 .0	.7		P 2	27						
6	49	C	TEH	TSH PS	BC028-08	A-720-M/ULC	02C 17.4	.7	141	1	34						
26	49	H	TEC	TEC	BH027-08	A-720-M/ULC	02C 15.7	.4	135	1	39						
45	49	C	TEH	TEH	BC026-07	A-720-M/ULC	AV4 .0	.5		P 2	22						
17	50	H	TEC	TEC PS	BH007-02	A-720-M/ULC	03H 25.0	.4	147	1	28						
34	51	C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV2 .0	1.3		P 2	30						
		C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV3 .0	.4		P 2	26						
34	53	C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV1 .0	.6		P 2	27						
		C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV2 .0	.6		P 2	26						
		C	TEH	TEH PS	BC021-06	A-720-M/ULC	AV3 .0	.5		P 2	26						
26	54	H	TEC	TEC PS	BH017-05	A-720-M/ULC	03C 40.3	.4	147	1	28						
29	55	H	TEC	TEC PS	BH017-05	A-720-M/ULC	01C 18.1	.5	155	1	20						
42	55	C	TEH	TEH PS	BC026-07	A-720-M/ULC	AV2 .0	.4		P 2	21						
		C	TEH	TEH	BC026-07	A-720-M/ULC	AV3 .0	1.0		P 2	26						
44	57	C	TEH	TEC SC	BH034-08	A-720-M/ULC	AV4 .0	1.2		P 3	24						
43	60	C	TEH	TEC PS	BH034-08	A-720-M/ULC	AV4 .0	.8		P 2	27						
35	66	C	TEH	TEH SS	BC029-08	A-720-M/ULC	06C .5	.6	124	P 1	27						
40	66	C	TEH	TEH PS	BC024-07	A-720-M/ULC	AV4 .0	.7	.	P 2	23						
39	69	C	TEH	TEC PS	BH034-08	A-720-M/ULC	AV2 .0	.6		P 2	26						
27	70	H	TEC	TEC SC	BH019-05	A-720-M/ULC	06H 4.8	.4	142	1	36						
29	70	H	TEC	TEC PS	BH019-05	A-720-M/ULC	05H 33.1	.3	136	1	37						
25	71	H	TEC	TEC SS	BH019-05	A-720-M/ULC	06H 3.0	.6	137	1	39						
		H	TEC	TEC PS	BH019-05	A-720-M/ULC	04C 37.5	.5	144	1	30						
11	72	H	TEC	TEC PS	BH009-03	A-720-M/ULC	02H 36.6	.6	154	1	22						

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G B
DESCRIPTION : 20% TO 39%

Date : 11/13/92
Time : 9:00 AM

			Extent				09/92						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
34	74	H	TEC	TEC PC	BH021-06	A-720-M/ULC	06C 5.4	1.2	155	1	20						
35	74	H	TEC	TEC PC	BH021-06	A-720-M/ULC	06C 5.6	.7	149	1	26						
28	75	H	TEC	TEC PS	BH020-06	A-720-M/ULC	04C 32.2	.4	151	1	26						
34	75	H	TEC	TEC PC	BH021-06	A-720-M/ULC	06C 5.5	.7	152	1	23						
35	75	H	TEC	TEC PC	BH021-06	A-720-M/ULC	02C .7	.5	140	1	34						
11	76	H	TEC	TEC	BH013-04	A-720-M/ULC	02H 36.0	.4	141	1	38						
11	78	H	TEC	TEC PS	BH013-04	A-720-M/ULC	01H 10.9	.4	156	1	23						
11	85	H	TEC	TEC PS	BH014-04	A-720-M/ULC	02H 36.5	.8	155	1	22						
11	89	H	TEC	TEC PS	BH015-04	A-720-M/ULC	02H 36.1	.5	138	1	34						

Number of RECORDS Selected from Current Outage : 95
Number of TUBES Selected from Current Outage : 81

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/92

COMPONENT : S/G C

Date : 11/16/92
Time : 11:30 AM

Examination Dates : 10/10/92 thru 10/18/92

Total Number of Tubes Inspected .....: 3188

Total Indications
Between 20% and 39% ....: 73
Greater than or equal to 40% ...: 3

Total Tubes Plugged as Preventive Maint : 2
Total Tubes Plugged ....: 5

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	2	TSC -.5 to 01C -2.1 :	3
01H -2.0 to 06H +2.0 :	6	01C -2.0 to 06C +2.0 :	6
06H +2.1 to AV1 -3.1 :	9	06C +2.1 to AV4 -3.1 :	13
AV1 -3.0 to AV4 -3.0 :	37		

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G C
DESCRIPTION : PLUGGABLE INDICATIONS

Date : 11/13/92
Time : 9:00 PM

Row	Col	Leg	Extent Req	Extent Tst/Note	Reel	Probe	09/92 Location	09/92 Volts	09/92 Deg	09/92 Ch	%	Diff	N/A Location	N/A Volts	N/A Deg	N/A Ch	N/A %
33	39	C	TEH	TEH *N	CC024-05	A-720-M/ULC	AV1 .0	2.3		35%	PTP						
35	41	C	TEH	TEH *N	CC024-05	A-720-M/ULC	AV1 .0	2.4		35%	PTP						
2	55	C	06C	06C PC	CC003-01	A-720-M/ULC	TSC 24.1	.3	105	1	60						
20	66	H	TEC	TEC PC	CH015-03	A-720-M/ULC	06C 2.4	.6	134	1	41						
2	70	C	06C	06C PC	CC003-01	A-720-M/ULC	02C .7	.5	107	1	45						

Number of RECORDS Selected from Current Outage : 5
Number of TUBES Selected from Current Outage : 5

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G C
DESCRIPTION : 20% TO 39%

Page : 1 of 2
Date : 11/13/92
Time : 9:00 AM

			Extent					09/92					N/A				
Row	Col	Leg	Req	Tst/Note	Reel'	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
3	10	H	06C	06C PS	CH001-01	A-700-SF/RM	02H 34.9	.9	142	1	34						
6	12	C	TEH	TEH PS	CC006-01	A-720-M/ULC	02C 11.1	.6	142	1	31						
26	15	C	TEH	TEH SS	CC017-03	A-720-M/ULC	05H 44.7	.8	141	1	33						
21	19	C	TEH	TEH SC	CC017-03	A-720-M/ULC	AV4 .3	.9		P 2	23						
37	28	C	TEH	TEH SS	CC023-05	A-720-M/ULC	AV4 .0	.6		P 3	24						
30	30	C	TEH	TEH SS	CC018-03	A-720-M/ULC	AV4 .0	.5		P 3	24						
39	30	C	TEH	TEH SS	CC023-05	A-720-M/ULC	AV2 18.8	1.3	152	1	25						
30	31	C	TEH	TEH PS	CC018-03	A-720-M/ULC	AV3 .0	.4		P 2	22						
33	31	C	TEH	TEH SS	CC023-05	A-720-M/ULC	AV3 .0	.7		P 3	24						
4	33	H	TEC	TEC	CH029-07	A-720-M/ULC	TSC 27.2	.5	142	1	33						
43	33	C	TEH	06C SS	CH034-09	A-720-M/ULC	AV3 .0	.6		P 2	23						
4	34	H	TEC	TEC SS	CH029-07	A-720-M/ULC	TSC 27.3	.7	147	1	27						
35	35	C	TEH	TEH PS	CC023-05	A-720-M/ULC	AV3 .0	.5		P 2	23						
35	36	C	TEH	TEH	CC024-05	A-720-M/ULC	AV2 .0	.5		P 2	23						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV3 .0	.5		P 2	23						
8	39	C	TEH	TEH PS	CC026-06	A-720-M/ULC	03C 48.3	.8	134	1	39						
		C	TEH	TEH PS	CC026-06	A-720-M/ULC	03C 11.0	.7	145	1	30						
33	39	C	TEH	TEH	CC024-05	A-720-M/ULC	AV1 .0	2.3		P 2	35						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV2 .0	.6		P 2	24						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV3 .0	.9		P 2	26						
34	41	C	TEH	TEH	CC024-05	A-720-M/ULC	AV1 .0	1.1		P 2	27						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV3 .0	.5		P 2	23						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV4 .0	1.0		P 2	26						
35	41	C	TEH	TEH	CC024-05	A-720-M/ULC	AV1 .0	2.4		P 2	35						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV2 .0	1.1		P 2	27						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV3 .0	.7		P 2	24						
		C	TEH	TEH	CC024-05	A-720-M/ULC	AV4 .0	.3		P 2	21						
33	43	C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV2 .0	.5		P 2	22						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV3 .0	.5		P 2	23						
35	43	C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV2 .0	.9		P 2	26						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV3 .0	1.4		P 2	29						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV4 .0	1.1		P 2	27						
13	44	H	TEC	TEH PS	CC031-06	A-720-M/ULC	02H 50.4	.7	148	1	29						
34	44	C	TEH	TEH SS	CC025-05	A-720-M/ULC	AV3 .0	.5		P 3	23						
35	44	C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV2 .0	1.6		P 2	30						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV3 .0	1.5		P 2	30						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV4 .0	.8		P 2	25						
23	45	C	TEH	TEH	CC020-04	A-720-M/ULC	AV3 .1	.6		P 2	24						
35	45	C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV2 .0	1.9		P 2	33						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV3 .0	.6		P 2	24						
		C	TEH	TEH PS	CC025-05	A-720-M/ULC	AV4 .0	.5		P 2	22						
4	46	C	TEH	TSH PS	CC026-06	A-720-M/ULC	TSH 28.4	.7	152	1	24						
30	46	C	TEH	TEH	CC020-04	A-720-M/ULC	AV1 .1	1.6		P 2	32						

CUMULATIVE EXAMINATION REPORT
PTN-3
OUTAGE : 09/92

COMPONENT : S/G C
DESCRIPTION : 20% TO 39%

Date : 11/13/92
Time : 9:00 AM

Row	Col	Leg	Extent Req	Extent Tst/Note	Reel	Probe	Location	09/92 Volts	09/92 Deg	09/92 Ch	%	Diff	N/A Location	Volts	Deg	Ch	%
		C	TEH	TEH	CC020-04	A-720-M/ULC	AV2 .1	1.7		P 2	32						
		C	TEH	TEH	CC020-04	A-720-M/ULC	AV3 .1	.9		P 2	26						
41	46	H	TEC	TEC	CH034-09	A-720-M/ULC	06H 5.8	.7	145	1	32						
30	48	H	TEC	TEC PS	CH018-04	A-720-SF/RM	AV2 .0	.8		P 2	28						
		H	TEC	TEC PS	CH018-04	A-720-SF/RM	AV3 .0	1.6		P 2	34						
35	49	H	TEC	TEC PS	CH030-07	A-720-M/ULC	AV4 .0	.6		P 2	23						
43	53	H	TEC	TEC	CH033-08	A-720-M/ULC	06C 3.2	.4	135	1	38						
39	54	H	TEC	TEC	CH031-07	A-720-M/ULC	AV3 .0	.3		P 2	20						
22	55	H	TEC	TEC SS	CH022-05	A-720-M/ULC	06H 2.2	.8	135	1	39						
26	58	H	TEC	TEC PS	CH022-05	A-720-M/ULC	AV2 .0	.6		P 2	22						
33	58	H	TEC	TEC PC	CH032-08	A-720-M/ULC	06C 37.3	.5	149	1	26						
30	61	H	TEC	TEC PS	CH022-05	A-720-M/ULC	AV2 .0	.7		P 2	24						
32	62	H	TEC	TEC PC	CH032-08	A-720-M/ULC	AV3 9.1	.4	145	1	30						
38	62	H	TEC	TEC PC	CH032-08	A-720-M/ULC	AV3 11.9	.5	145	1	30						
24	63	H	TEC	TEC SC	CH023-05	A-720-SF/RM	AV2 .0	.5		P 2	24						
		H	TEC	TEC SC	CH023-05	A-720-SF/RM	AV3 .0	.4		P 2	23						
7	64	H	TEC	TEC PC	CH009-02	A-720-SF/RM	06H 14.1	.6	151	1	25						
20	64	H	TEC	TEC	CH014-03	A-720-M/ULC	01C 50.3	.7	156	1	24						
30	64	H	TEC	TEC SC	CH023-05	A-720-SF/RM	06H 2.3	.6	159	1	20						
32	64	H	TEC	TEC	CH032-08	A-720-M/ULC	02H -.6	.6	106	P 1	37						
38	65	H	TEC	TEC PS	CH032-08	A-720-M/ULC	AV2 .0	.4		P 2	23						
		H	TEC	TEC PS	CH032-08	A-720-M/ULC	AV3 .0	.3		P 2	21						
		H	TEC	TEC PS	CH032-08	A-720-M/ULC	AV4 .0	.7		P 2	24						
38	71	H	TEC	TEC PS	CH026-06	A-720-M/ULC	AV3 .1	.7		P 2	21						
35	72	H	TEC	TEC PS	CH026-06	A-720-M/ULC	AV2 7.9	.5	138	1	35						
32	75	H	TEC	TEC PS	CH026-06	A-720-M/ULC	06H 22.8	.3	149	1	27						
1	76	C	06C	06C	CC004-01	A-700-SF/RM	02C 4.8	1.3	139	1	35						
13	84	H	TEC	TEC PS	CH017-04	A-720-SF/RM	04H 40.4	.6	157	1	23						
5	87	H	TEC	TEC PS	CH011-02	A-720-M/ULC	02H 15.4	.5	156	1	20						
3	88	H	06C	06C PC	CH007-01	A-680-SF/RM	BAH 19.9	.4	150	1	28						

Number of RECORDS Selected from Current Outage : 73
Number of TUBES Selected from Current Outage : 51

FPL

FORM NIS-BB OWNERS' DATA REPORT FOR EDDY CURRENT EXAMINATION RESULTS

As required by the provisions of the ASME CODE RULES

EDDY CURRENT EXAMINATION RESULTS

PLANT : Turkey Point Unit 4

EXAMINATION DATES: APRIL 24, 1993 thru APRIL 28, 1993

STEAM GENERATOR	TOTAL TUBES INSPECTED	TOTAL TUBES 20% - 39%	TOTAL TUBES 40% - 100%	TUBES PLUGGED AS PREVENTIVE MAINTENANCE	TUBES PLUGGED THIS OUTAGE	TOTAL PLUGGED TUBES IN S/G
4E210A	3198	2	NONE	NONE	NONE	16
4E210B	3206	9	NONE	NONE	NONE	8
4E210C	3205	19	NONE	NONE	NONE	9

LOCATION OF INDICATIONS
(20% - 100%)

STEAM GENERATOR	AVB BARS	SUPPORT LOCATIONS 1 THROUGH 6		TOP OF TUBE SHEET TO #1 SUPPORT		TOTAL INDICATIONS	
		COLD LEG	HOT LEG	COLD LEG	HOT LEG	20% - 39%	40% TO 100%
4E210A	NONE	2	1	NONE	NONE	3	NONE
4E210B	3	1	2	1	2	9	NONE
4E210C	4	12	7	NONE	NONE	23	NONE

Remarks:

CERTIFICATION OF RECORD

We certify that the statements in this report are correct and the tubes inspected were tested in accordance with the requirements of Section XI of the ASME Code.

Florida Power & Light Co.

DATE: 6-4-93 PREPARED BY: [signature]
S/G EDDY CURRENT COORDINATOR

DATE: 6-4-93 REVIEWED BY: [signature]
INSPECTIONS SUPERVISOR

DATE: 6-14-93 APPROVED BY: K R Craig
S/G PROGRAM MANAGER

forms II/nisbbsl2.drw

an FPL Group company

CUMULATIVE EXAMINATION REPORT
PTN-4
OUTAGE : 04/93

COMPONENT : S/G A
DESCRIPTION : 20% TO 100%

Date : 6/ 4/93
Time : 9:35 AM

			Extent				04/93						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
28	14	C	TEH	TEH	AC006-02	A-720-M/ULC	01H 42.6	1.6	145	1	32						
		C	TEH	TEH	AC006-02	A-720-M/ULC	02C 2.7	.8	146	1	31						
14	82	H	TEC	TEC PS	AH004-02	A-720-M/ULC	04C 9.5	.8	157	1	27						

Number of RECORDS Selected from Current Outage : 3
Number of TUBES Selected from Current Outage : 2

CUMULATIVE EXAMINATION REPORT
PTN-4
OUTAGE : 04/93

COMPONENT : S/G B
DESCRIPTION : 20% TO 100%

Page : 1 of 1
Date : 6/ 4/93
Time : 9:35 AM

			Extent				04/93						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
8	18	C	TEH	TEH	BC016-04	A-700-M/ULC	TSC 3.9	.7	148	1	30						
3	24	C	06C	06C	BC011-03	A-720-M/ULC	03C 25.5	.9	150	1	28						
22	37	H	TEC	TEC SS	BH026-07	A-720-M/ULC	02H .0	.4	143	1	32						
45	42	H	TEC	TEC	BH013-03	A-720-M/ULC	AV2 .0	.5		P 2	22						
45	43	H	TEC	TEC	BH013-03	A-720-M/ULC	AV2 .0	.6		P 2	23						
22	48	H	TEC	TEC PC	BH023-06	A-720-M/ULC	TSH 6.3	.7	143	1	32						
45	48	H	TEC	TEC PS	BH011-03	A-720-M/ULC	AV4 .0	.7		P 2	24						
37	69	H	TEC	TEC SS	BH030-03	A-720-M/ULC	TSH 21.6	.7	145	1	31						
14	82	H	TEC	TEC SS	BH004-02	A-720-M/ULC	02H 15.6	1.2	150	1	26						

Number of RECORDS Selected from Current Outage : 9
Number of TUBES Selected from Current Outage : 9

CUMULATIVE EXAMINATION REPORT
PTN-4
OUTAGE : 04/93

COMPONENT : S/G C
DESCRIPTION : 20% TO 100%

Date : 6/ 4/93
Time : 9:35 AM

			Extent				04/93						N/A				
Row	Col	Leg	Req	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
28	28	C	TEH	TEH	CC016-04	A-720-M/ULC	05H 44.0	.9	153	1	26						
		C	TEH	TEH PS	CC016-04	A-720-M/ULC	06H -.9	1.0	149	1	30						
37	32	C	TEH	TEH	CC018-04	A-720-M/ULC	06C -.7	.7	119	P 1	37						
26	37	C	TEH	TEH	CC014-03	A-720-M/ULC	05C 31.5	.8	157	1	20						
3	52	C	06C	06C PS	CC013-03	A-720-M/ULC	01C 29.6	.6	152	1	25						
43	52	H	TEC	TEC SS	CH022-05	A-720-M/ULC	05H 12.0	.6	150	1	24						
40	53	C	TEH	TEH SC	CC021-05	A-720-M/ULC	06C -.5	.6	127	P 1	30						
24	56	H	TEC	TEC PS	CH023-06	A-720-SF/RM	03C 38.7	.6	135	1	25						
42	56	H	TEC	TEC SS	CH022-05	A-720-M/ULC	06C -.6	1.2	120	P 1	24						
		H	TEC	TEC SS	CH022-05	A-720-M/ULC	05C 32.3	.5	141	1	31						
		H	TEC	TEC SS	CH022-05	A-720-M/ULC	05C 13.2	.6	143	1	30						
33	61	C	TEH	TEH SS	CC021-05	A-720-M/ULC	03C 26.1	.3	144	1	33						
24	62	H	TEC	TEC PS	CH024-06	A-720-M/ULC	02C 35.3	.6	162	1	21						
37	69	C	TEH	TEH PC	CC022-06	A-720-M/ULC	06H .1	.5	150	1	25						
32	70	H	TEC	TEC PS	CH008-03	A-720-M/ULC	AV1 .0	.5		P 2	22						
16	72	H	TEC	TEC	CH002-01	A-720-M/ULC	05H 43.0	.4	147	1	30						
30	72	H	TEC	TEC SC	CH006-02	A-720-M/ULC	04C 21.1	.3	152	1	26						
37	72	H	TEC	TEC PS	CH008-03	A-720-M/ULC	05H 45.0	.5	139	1	36						
		H	TEC	TEC	CH008-03	A-720-M/ULC	AV3 -.2	.5	148	P 2	23						
23	77	H	TEC	TEC PS	CH007-02	A-720-M/ULC	02C 50.6	.4	148	1	28						
31	80	H	TEC	TEC SS	CH022-05	A-720-M/ULC	06H 2.3	.4	142	1	31						
27	81	H	TEC	TEC SC	CH007-02	A-720-M/ULC	AV1 .0	.3		P 2	20						
26	82	H	TEC	TEC SC	CH007-02	A-720-M/ULC	AV1 .0	.3		P 2	20						

Number of RECORDS Selected from Current Outage : 23
Number of TUBES Selected from Current Outage : 19

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 4
04/93

COMPONENT : S/G A

Date : 06/14/93
Time : 1:30 PM

Examination Dates : 04/24/93 thru 04/28/93

Total Number of Tubes Inspected .....: 3198

Total Indications
- Between 20% and 39% ....: 3
- Greater than or equal to 40% ...: 0

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 16

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	0	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	1	01C -2.0 to 06C +2.0 :	2
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	0		

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 4
04/93

COMPONENT : S/G B

Page : 1 of 1
Date : 06/14/93
Time : 1:30 PM

Examination Dates : 04/24/93 thru 04/28/93

Total Number of Tubes Inspected .....: 3206

Total Indications
Between 20% and 39% ....: 9
Greater than or equal to 40% ...: 0

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 8

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	2	TSC -.5 to 01C -2.1 :	1
01H -2.0 to 06H +2.0 :	2	01C -2.0 to 06C +2.0 :	1
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	1
AV1 -3.0 to AV4 -3.0 :	2		

CUMMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 4
04/93

COMPONENT : S/G C

Page : 1 of 1
Date : 06/14/93
Time : 1:30 PM

Examination Dates : 04/24/93 thru 04/28/93

Total Number of Tubes Inspected .....: 3205

Total Indications
    Between 20% and 39% ....: 23
    Greater than or equal to 40% ...: 0

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 9

Location Of Indications 20% to 100%

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	0	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	6	01C -2.0 to 06C +2.0 :	12
06H +2.1 to AV1 -3.1 :	4	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	1		