TURKEY POINT PLANT UNITS 3 AND 4

DOCKET NUMBERS 50-250 AND 50-251

CHANGES, TESTS, AND EXPERIMENTS
MADE WITHOUT PRIOR COMMISSION APPROVAL

FOR PERIOD

JULY 1, 1986 THROUGH JUNE 30, 1987

IN COMPLIANCE WITH
TITLE 10, SECTION 50.59(b)
CODE OF FEDERAL REGULATIONS

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INTRODUCTION

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

i) changes in the facility as described in the FSAR
ii) changes in the procedures as described in the FSAR, and
iii) tests and experiments not described in the FSAR

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 July 1, 1986 through June 30, 1987.

This report is divided into three sections; the first, Plant Change/Modifications, covering changes in the facility as described in the FSAR; the second, Procedure Changes covering changes in the procedures as described in the FSAR; and the third, Tests and Experiments, covering tests and experiments not described in the FSAR.

Appendix A to this report is a list of safety and power operated relief valve actuations, which is submitted in accordance with FPL's commitment to comply with the requirements of Item IIK.3.3 of NUREG 0737. This report covers the period from July 1, 1986 to June 30, 1987.

Appendix B to this report is a summary of the findings of the Steam Generator tube inspection performed on Unit 3 during the report period from July 1, 1986 through June 30, 1987.

TITLE 10, SECTION 50.59 CFR REPORT

i) COMPLETED PC/M LIST

JULY 1, 1986 THRU JUNE 30, 1987

PC/M	TITLE	UNIT	TURNED OVER DATE
85-86	AFW CV INSTRUMENT AIR FILTER MODIFICATION	4	7/26/86
85-108	NUCLEAR ADMIN. BLDG. POWER SUPPLY	3 & 4	3/15/86
85-171	INSTALLATION OF CHAIN OPERATOR ON AFW ISOLATION VALVES	4	7/26/86
85-196	DIESEL GENERATOR SKID TANK SOLENOID VALVE BYPASS LINE ADDITION	3 & 4	8/5/86
86-62	NORMAL CONTAINMENT COOLING FAN MODIF.	4	8/1/86
86-94	EMERGENCY DIESEL GENERATOR - "B" ROOM VENT FAN 4V34 POWER SUPPLY	4	7/27/86
86-95	PROVIDE NEW POWER FEED FOR NON-VITAL SECTION OF MCC 3A & 3D	3	7/24/86
86-041	MODIF. OF MCC "D" AUTO TRANSFER	3	7/25/86
78-102B	STEAM GENERATOR BLOWDOWN RECOVERY SYSTEM	4	7/17/86
84-02	MODIF. TO COMPLY WITH REG.GUIDE 1.97 REV. 3 REQUIREMENTS TO PROVIDE QUALIFIED LIMIT SWITCHES	4	9/15/86
83-63	IMPROVED FLOOR DRAIN FOR CONTAINMENT SPRAY PUMP ROOM	3	12/12/85
83-114	REACTOR CAVITY FILTERS - LEAD SHIELDING	4	4/4/86
86-03	MSIV NITROGEN SUPPLY ADDITION	4	8/7/86
86-64	4.160KV FUSE HOLDER SUBSTITUTION	3 & 4	8/7/86
86-70	REPAIR DAMAGED T/C CABLES AT TE-4-1418 & 1421	4	6/24/86
86-71	ICW BASKET STRAINER REPLACEMENT	4	6/24/86
82-83	ADD BACK-UP TO RELAYING IN SWITCHYARD	3 & 4	7/24/85
86-44	ICW BASKET STRAINER BELZONA LINING	4	12/1/86
86-87	CABLE REPLACEMENT FOR MSIV SOLENOID	4	12/31/86
85-152	NIS INPUT TO TURBINE RUNBACK - REINSTATE 1/4 CONFIGURATION	4	12/18/86
85-85	AFW CV INSTRUMENT AIR FILTRATION MODIF.	3	9/30/86
83/139	HALON SUPPRESION SYSTEM FOR APPENDIX R	3 & 4	8/20/86

TITLE 10, SECTION 50.59 CFR REPORT
i) COMPLETED PC/M LIST
JULY 1, 1986 THRU JUNE 30, 1987

PC/M	TITLE	UNIT	TURNED OVER DATE
85-56	REPLACEMENT OF INTAKE WALKWAY	3 & 4	10/31/86
84-210	TURBINE RUNBACK MODIFICATION	4	8/19/86
85-151	ICW PIPE & STRAINER INSPECTION CLEANING	4	8/27/86
85-124	REMOVAL OF SPENT RESIN PIPE IN LAUNDRY ROOM	3 & 4	10/31/86
86-22	ICW HEADER ISOLATION VALVE REPLACEMENT	4	9/8/86
84-124	AFW FLOW TRANSMITTER REPLACEMENT	4	3/9/87
84-158	EMERGENCY DIESEL GENERATOR "4B" CFD RELAY REPLACEMENT	4	12/17/85
83-141	FIRE BARRIERS FOR APPENDIX R	3	5/21/86
83-145	FIRE DAMPERS FOR APPENDIX R	3 & 4	8/14/86
84-20	POST ACCIDENT CONTAINMENT AIR SAMPLING SYSTEM FLOW TRANSMITTER	3 & 4	4/25/86
86-067	TURBINE AUX. BLOCKING OF AUTO LOAD ON DIESEL GENERATORS	4	3/9/87
86-05	MSIV NITROGEN SUPPLY ADDITION (INTERIM)	3	6/9/86
85-130	AFW DISCHARGE FLOW CONTROL VALVE UPGRADE	3	3/9/87
85-149	SPENT FUEL PIT AIR INLET DAMPER REPLACEMENT	3	3/9/87
82-36	SPENT FUEL PIT LEVEL INDICATOR AND ALARM	4	5/15/86
84-167	DECONTAMINATION SHOWER FACILITY	3 & 4	3/26/87
86-158	THROWBOLT REPLACEMENT ON INTAKE STRUCTURE BAY HATCHWAY	3 & 4	11/18/86
86-166	FUEL TRANSFER SYSTEM CABLE DRIVE MODIF. DISSASSEMBLY OF PRESENT SYSTEM	3	3/17/87
86-207	IN SERVICE TESTING GAUGE INSTALLATION FOR SPENT FUEL PIT COOLING PUMPS	3	3/19/87
86-159	INTAKE STRUCTURE WOOD GRATING LATCHES	3 & 4	11/8/86
86-124	REPLACEMENT HIGH RANGE GAMMA RADIATION READOUT MODULE	4	11/26/86
83-140	FIRE DETECTION FOR APPENDIX "R" MODIFICATION	3 & 4	12/1/86
84-124	AFW FLOW TRANSMITTER REPLACEMENT	4	3/9/87
82-311	AFW TURBINE STEAM SUPPLY STOP/CHECK VALVE	3	3/13/87
85-143	BREAKER/FUSE COORDINATION MODIF.	3 & 4	6/13/87

TITLE 10, SECTION 50.59 CFR REPORT
i) COMPLETED PC/M LIST
JULY 1, 1986 THRU JUNE 30, 1987

PC/M	TITLE	UNIT	TURNED OVER DATE
85-60	MAIN TRANSFORMER FAN COOLER UPGRADE	3	6/11/87
84-171	MODIFICATION TO ALLEVIATE SHORTAGE OF COMPARTMENTS ON MCC "3B"	3	2/2/87
86-236	REPLACEMENT OF TREATED WATER PUMP SEALS	3 & 4	4/4/87
85-197	RELOCATE INSTRUMENT AIR SUPPLY VALVES 40-4-098 & 40-3-641	3	9/26/86
85-170	INSTALLATION OF AFW VALVE ACCESS PLATFORM	3	1/15/87
86-67	TURBINE AUXILIARIES-BLOCKING OF AUTO-LOADING ON DIESEL GENERATOR	4	3/9/87
87-086	PERSONNEL AIRLOCK EQUALIZATION VALVE REPLACEMENT	4	3/28/87
87-095	"4A" ICW PUMP ANCHOR BOLT REPLACEMENT	4	5/14/87
84-121	UPENDER LEVELING DEVICE MODIF.	3	3/30/87
86-029	AFW LOCAL INDICATION UNDER THE MAIN FEED WATER PLATFORM	3 & 4	3/28/87
86-021	"C" AUX. FEEDPUMP REPLACEMENT IMPELLER	3 & 4	4/13/87
86-212	ENVIRONMENTAL QUALIFICATION LIST REVISION	3 & 4	2/12/87
86-103	ENVIRONMENTAL QUALIFICATION LIST REVISION	3 & 4	2/12/87
85-65	G. E. SAM RELAY MODIFICATION, PC CARD REPLACEMENT	4	1/15/87
83-153	CABLE REROUTING - APPENDIX R MODIFICATION	3 & 4	12/18/86
87-23	MAIN STEAM HYDRAULIC SNUBBER REPLACEMENT	4	6/19/87
86-96	NEW POWER FEED TO NON-VITAL SECTION OF MCC "4A"	4	9/8/86
86-60	COMPUTER ROOM TEMPERATURE INDICATION	3 & 4	2/20/87
85-071	SPENT FUEL PIT BUILDING WALL JOINT REPAIR	3	4/15/87
85-35	REPLACEMENT OF PYCO RTD'S	3	6/19/87
87-97	INSTALLATION OF UNDER VOLTAGE TRIP CIRCUITRY FOR TURBINE AND POLAR CRANE BREAKERS	3	6/27/87

TITLE 10, SECTION 50.59 CFR REPORT
i) COMPLETED PC/M LIST
JULY 1, 1986 THRU JUNE 30, 1987

PC/M	TITLE	UNIT	TURNED OVER DATE
87-52	GENERATOR NEUTRAL GROUNDING TRANSFORMER REPLACEMENT	3	6/9/87
87-53	GENERATOR NEUTRAL GROUNDING TRANSFORMER REPLACEMENT	4	6/9/87
87-98	INSTALLATION OF UNDERVOLTAGE TRIP DEVICE FOR POLAR CRANE BREAKERS	4	6/23/87
80-117	UPGRADE AFW SUCTION, DISCHARGE, STEAM SUPPLY	3 & 4	11/15/85
84-111	A/C UNIT FOR PASS CONTROL PANEL	3 & 4	10/31/86
86-15	REPLACED TELEDYNE-FARRIS COMPONENT COOLING WATER RELIEF VALVES	4	10/13/86
86-68	REMOVAL OF CCW PIPING TO THE PRIMARY SHIELD COOLERS	4	10/31/86
87-169	MODIFICATION TO COMPONENT COOLING WATER SYSTEM	4	6/19/87
81-59	WATER TREATMENT PLANT FINAL EFFLUENT CONDUCTIVITY TRIP	3 & 4	2/17/87
83-209	MSR FOUR TUBE PASS MODIFICATION	4	11/18/86
85-131	AUXILIARY DISCHARGE FCV UPGRADE	4	9/17/86
85-133	MSR MODERNIZATION	4	11/18/86
86-31	AFW PUMP CONTROL PANEL WIRING MODIFICATION	3 & 4	12/10/86
87-99	ICW/CCW BASKET STRAINER REPLACEMENT	3	6/26/87
87-156	ICW BASKET STRAINER ISOLATION VALVE REPLACEMENT-SHAFT/OPERATOR ADAPTER	3	6/16/87
86-80	SAFETY INJECTION ACCUMULATOR MAKE UP HEADER SEISMIC REPLACEMENT	3	6/26/87
86-76	DIESEL GEN "B" FREQUENCY METER REPLACEMENT	3 & 4	6/3/87
86-90	ROOT VALVE # 4-20-698 SUBSTITUTION	4	7/8/86
85-10	ADDITION OF FW CONTROL VALVE DIRECT POSITION INDICATION	4	2/18/87
84-209	REINSTATEMENT OF POWER MISMATCH WITHOUT ROD WITHDRAWAL	4	8/16/86

TITLE 10, SECTION 50.59 CFR REPORT
i) COMPLETED PC/M LIST
JULY 1, 1986 THRU JUNE 30, 1987

PC/M	TITLE	UNIT	TURNED OVER DATE
86-162	REMOVAL OF CCW PIPING TO THE PRIMARY SHIELD COOLERS	3	5/15/87
87-126	ACCUMULATOR SAFETY INJECTION TEST LINE SOLENOID VALVE REPLACEMENT	4	7/13/87
87-160	BAILEY TEMP. TRANSMITTER REPLACEMENT FOR TURBINE PLANT COOLING WATER	3	7/13/87
87-037	ICW PUMP FOUNDATION REPAIR ANCHOR BOLT REPLACEMENT	3	6/9/87
87-102	REACTOR VESSEL HEAD INSULATION-REFLECTIVE REPLACEMENT	4	6/9/87
87-101	REACTOR VESSEL HEAD INSULATION-PERMANENT REPLACEMENT	4	6/9/87
87-177	CONTAINMENT SPRAY RESTRICTING ORIFICE	4	6/11/87
86-100	NIS SOURCE RANGE PRE-AMPIFIERS	4	10/31/86
86-184	RPI INVERTER REGULATOR TRANSFORMER REPLACEMENT	4	5/30/87
83-50	MASONRY WALL MODIFICATION	3 & 4	2/12/87
87-210	RPLACMENT SUPPORTS 4-SIH-42 AND 4-PRWH-11	4	6/12/87
84-16	RHR ISOLATION VALVE CIRCUIT MODIFICATION	3	4/13/87
85-182	CHEMICAL ADDITION LINES SUPPORT REPAIR	3	10/29/86
85-139	REMOVE VALVES 4-524/525 AND PIPING	4	9/5/86
85-181	ROMOVE INACTIVE NITROGEN BLANKET LINE	4	10/3/86
84-11	MODIFICATION TO PRESSURIZER SPRAY SYSTEM (I.C.)	3	7/3/85
85-141	FUEL TRANSFER SYSTEM MANIPULATOR CRANE DUAL CABLE MODIFICATION	3	6/29/87
86-121	CONTROL POWER FUSE REPLACEMENT	4	5/30/87
86-107	MCC CONTROL POWER FUSE REPLACEMENT	3	5/30/87
86-026	4KV SWITCHGEAR BREAKER ELEVATING MECHANISM REBUILDING	4	5/30/87
85-11	MODIFICATION OF 4160V BREAKER "HH" SWITCHES	3	6/3/87
85-149	SPENT FUEL PIT INLET DAMPER REPLACEMENT	3	5/7/87

PLANT CHANGE/MODIFICATION 85-086

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 07/26/86

SUMMARY DATE: 09/03/86

REVISION: 0

AUXILIARY FEEDWATER CV INSTRUMENT AIR FILTRATION MODIFICATION

Summary:

This modification provided for the installation of new filters in the instrument air (nitrogen backed) supply line to each unit 4 Auxiliary Feedwater Control Valve as shown on the drawing listed on Attachment 1. The installation of the filters provides better quality air to the Control Valve positioners and actuators.

The new filters were installed upstream of the instrument air-nitrogen supply tie-in connections in the instrument air supply line. The filters are provided with isolation valves and bypass lines with a valve for ease of maintenance without isolating the Control Valve air supply. A new anchor, and, supports as required per 5177-PS-21 must be installed to isolate the filter assembly from the downstream check valve. This isolation is required to ensure the functional and structural integrity of the safety related portion of the system. All new filters, valves, and piping upstream of the anchor need not be supported to seismic Category I requirements. The new piping will be 1/2-inch galvanized steel, Schedule 40.

The new filters are Parker Hannifin Standard Airline Filters designed to separate dirt particles and water. Sizing for this installation was determined by the maximum flow capacity required for operation of the positioners on the Auxliliary Feedwater Control Valves. Using the flow rates required with 100 psi instrument air and 60 psi instrument air as the pressure range to operate the positioner, (13 scfm and 8.5 scfm respectively), and a required 2 to 5 psi pressure drop across the filter for efficient operation, and the particle size filtration needed, in this case 5 micron, a filter size selection is made form a chart supplied by the filter manufacturter. Therefore, the filter size selected is specific for this application, and has the concurrence of the filter manufacturer. This analysis is documented in Calculation M08-0424-01.

The method of operation of the air filter is such that pressurized air (instrument air at 100 psig) flows through a louvred deflector and is directed into a swirling pattern. Liquids and large dirt particles are thrown against the inside wall of the see-through polycarbonate bowl by cyclonic action and fall into a quiet zone below the lower baffle. This lower baffle maintains the quiet zone to prevent turbulent air from returning liquids and solids into the air stream. The instrument air, now free of liquids and dirt particles, pass through a filter element sized to remove particles down to 5 microns for this application. Clean air then flows through the outlet port. Liquids are discharged from the bowl by the automatic drain valve. The Parker Hannifin model number for this installation is 04F15B which translates to the following:

U4F - Minature Series
1 - 1/4" NPTF Port Size
5 - Automatic Drain
B - 5 Micron Filtering Element

Safety Evaluation:

These modifications provide for the installation of new filters in the instrument air supply line to each Unit 4 Auxiliary Feedwater Flow Control Valve. The air filters, their associated isolation and bypass valves and piping will not be installed to the requirements of seismic Category I. The seismic boundary anchor assures the structural integrity of the piping and components downstream of the anchor. The modifications provided by the PC/M include passive components whose function will not be impaired by any design basis accident described in the FSAR.

These modifications do not introduce new safety related equipment which could be affected by fire or add new combustibles which could invalidate the Fire Zone Heat Loading Analysis previously submitted to FPL per Bechtel letter SFB-1741 dated April 24, 1985. In addition, these modifications to not adversely affect any existing or proposed fire protection features of the plant. Therefore, these modifications do not affect the Turkey Point Fire Protection Program.

These modifications are not inside containment, are not attached to block walls, do not involve safety related snubbers and do not impact spent fuel pool cooling operations of the plant.

No special ALARA considerations are required because the modifications are to be carried out in the areas outside of Radiation Control Area.

The modification does not involve the addition of electrical cable or any changes to existing raceways. The final modifications accomplished by this PC/M do not affect the flooding analysis as described in the NRC Safety Evaluation Report dated September 4, 1979, because they do not introduce a new source of flooding, modify the existing flood mitigating features, or install or modify any safety related components which could be affected by flooding.

Based on the preceeding, the following conclusions can be made:

- The probability of occurrence of an accident previously evaluated in the FSAR will not be increased because the modifications do not alter the function of Instrument Air or Nitrogen Backup Supply to AFW Control Valves.

- The consequences of an accident previously evaluated in the FSAR will not be increased because the added tubing, filters and valves will not affect the performance of safety features.

- The changes which are associated with plant safety features are minor in nature and do not change the function of the plant safety features. Therefore, there is no possibility that an accident may be created that is different than any already evaluated in the FSAR.

- This probability of occurrence of an equipment malfunction important to safety which have already been evaluated in the FSAR, will not be increased due to the installation of filters and isolation valves because this change does not adversely affect any equipment important to safety.

- The consequences of equipment malfunction important to safety which have already been evaluated in the FSAR will not be increased because the performance design basis has nt been changed from that described in the FSAR.

- The modification to the AFW Control Valve Instrument Air Supply System identified in this PC/M will not change the inherent function or design basis for the system. In addition, the in-line air filters will be inspected and cleaned (if necessary) on a regular basis in accordance with approved system maintenance procedures, to prevent possible blockage. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than previously evaluated in the FSAR will not be created.

- This modification provides cleaner air to the AFW Control Valve actuator and thus does not reduce the margin of safety as defined in the bases for any Technical Specification.

Based on the above, these modifications do not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-108

PC/M CLASSIFICATION: NNS

UNIT: 3 and 4

TURNED OVER DATE: 3/15/86

SUMMARY DATE: 9/02/86

REVISION: 0

NUCLEAR ADMINISTRATION BUILDING POWER SUPPLY

Summary:

This PC/M installed a power supply from the existing Florida City Feeder to the new Nuclear Administration Building.

Safety Evaluation:

This PC/M does not adversely affect the operation of any nuclear safety related equipment as the Florida City Feeder and the Nuclear Administration Building do not perform a nuclear safety function and all work associated with the installation of this PC/M is outside the plant perimeter fence. Therefore, this PC/M is classified as non-nuclear safety related and does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-171

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 07/26/86

SUMMARY DATE: 09/02/86

REVISION: 0

INSTALLATION OF CHAIN OPERATORS ON AUXILIARY FEEDWATER ISOLATION VALVES

Summary:

This modification provided for the addition of chain operators to AFW manual isolation valves which are currently inaccessible. This PC/M has been reviewed with respect to the reference documents.

All applicable design verification elements of EDPI 3.16-10, Exhibit H were considered during the review.

Safety Evaluation:

This modification installs chain operators for operation of inaccessible AFW manual isolation valves that require local operator action under off-normal operating conditions. The effect of the addition of the chain operator on the pipe stress analysis and the design of the pipe supports has been evaluated and no modifications are required. The valve manufacturer has evaluated the addition of the chainwheel to the valve and has concluded that the seismic analysis remains valid (Refer to Attachment 7).

This modification is not inside containment, does not involve safety related snubbers, does not involve block walls, does not impact the spent fuel cooling operations of the plant, does not affect the Radioactive Waste Treatmen System of the plant, and does not involve the addition of electrical cable or changes to existing raceways.

The modification has been reviewed for ALARA requirements based upon the guidances provided in Criteria for ALARA Evaluation per FPL letter JPE-PTPO-84-1239 and is acceptable.

The modificatior accomplished by this PC/M does not affect the flooding analysis as described in the NRC Safety Evaluation Report, dated September 4, 1979, because the modification does not introduce a new source of flooding, modify the existing flood mitigating features, or install any safety related components which could be affected by flooding.

Based on the preceeding, the following conclusions can be made:

- The addition of chain operators does not change the inherent function of the AFW System. The pipe and pipe support design have been evaluated with respect to this change and the additional weight is acceptable and no pipe support changes are required. Therefore, the probability of occurrence of an accident previously evaluated in the FSAR will not be increased.

- The consequences of an accident previously evaluated in the FSAR will not increase because the modification does not affect the operation of any safety related system.

- This modification does not change the function of any safety related system. Therefore, there is no possibility that an accident may be created that is a different type than any previously evaluated in the FSAR.

- This modification does not affect the probability of occurrence of any equipment malfunction important to safety previously evaluated in the FSAR, because it does not adversely affect the inherent function, operation, or availability of equipment important to safety.

- Due to this modification, consequences of equipment malfunction important to safety is not changed from one which is evaluated in the FSAR because this modification does not change the design basis described in the FSAR for any safety related system.

- These modifications do not change the inherent function or design basis of the systems related to safety; therefore, the possibility of a malfunction of equipment important to safety which is of a different type than previously evaluated in the FSAR will not be created.

- This modification does not change the operation of any system related to safety, therefore, this modification does not reduce the margin of safety as defined in the bases for any Technical Specification.

Based on the above, this modification does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-196 PC/M CLASSIFICATION: NS

UNIT: 3 & 4

TURNED OVER DATE: 08/05/86

SUMMARY DATE: 09/02/86

RE ISION: 0

DIESEL GENERATOR SKID TANK SOLENOID VALVE BYPASS LINE ADDITION

Summary:

Add bypass lines with manual isolation valves for Solenoid Valves SV-3-3522 and SV-4-3522 in the respective fuel oil supply line to the Emergency Diesel Generator Skid Tanks. The installation of the bypass line provides an alternate fill path should the solenoid valve fail closed as a result of loss of power or a spurious signal to this solenoid valve. This modification is necessary to support the Appendix R Safe Shutdown System Analysis.

Safety Evaluation:

This modification provides bypass lines for Valves SV-3-3522 and SV-4-3522, fuel oil supply lines to the EDG Skid Tanks, to support licensing commitments associated with 10CFR50 Appendix R requirements. The new pipe will be seismically supported in accordance with project requirements for small pipe. Since the seismic integrity of the entire line, including this modification, will be evaluated and ensured under the program to walkdown and evaluate small piping and tubing, this approach is acceptable.

This modification is not inside containment, does not involve safety related snubbers, does not involve block walls, does not impact the spent fuel cooling operations of the plant, does not affect the Radioactive Waste Treatment System of the plant, and does not involve the addition of electrical cable or changes to existing raceways.

The modification has been reviewed for ALARA requirements based upon the guidances provided in Criteria for ALARA Evaluation per FPL letter JPE-PTPO-84-1239 and is acceptable.

The modification accomplished by this PC/M does not affect the flooding analysis as described in the NRC Safety Evaluation Report, dated September 4, 1979, because the modification does not introduce a new source of flooding, modify the existing flood mitigating features, or install any safety related components which could be affected by flooding.

Based on the preceeding, the following conclusions can be made:

° The probability of occurrence of an accident previously evaluated in the FSAR will not be increased because the modification to the fuel line pressure boundary is being performed to the same standards of the original pressure boundary and does not alter the function of the Emergency Diesel Generator System.

° The consequences of an accident previously evaluated in the FSAR will not increase because the added piping will not affect the performance of safety features.

° The changes are minor in nature and do not change the inherent function of any plant safety features system. Administrative procedures shall ensure that the bypass valve is kept closed during normal operation. In the event that the bypass valve is left open, the "Diesel Engine Trouble" alarm will sound when the fuel level in the Skid Tank reaches 4" from the top. Therefore, there is no possibility that an accident may be created that is different than any already evaluated in the FSAR.

° This modification does not affect the probability of occurrence of any equipment malfunction important to safety previously evaluated in the FSAR, because it does not adversely affect any equipment important to safety.

° Due to this modification, consequences of equipment malfunction important to safety is not changed from one which is evaluated in the FSAR because this modification does not change the design basis described in the FSAR for any safety related system.

° The modification to the Emergency Diesel Generator System identified in this PC/M will not change the inherent function or design basis for the system. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than previously evaluated in the FSAR will not be created.

° This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. It enhances the availability of a safety related system in the event of a fire.

Based on the above, these modifications do not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-062

PC/M CLASSIFICATION:	NS
UNIT:	4
TURNED OVER DATE:	08/01/86
SUMMARY DATE:	09/03/86
REVISION:	0

NORMAL CONTAINMENT COOLING FAN MODIFICATIONS

Summary:

This design package provides modifications in the control circuits of the Normal Containment Cooling (NCC) Fans 4V1A, 4V1B, 4V1C and 4V1D to prevent automatic loading of any of these fans on the Emergency Diesel Generator (EDG) on restoration of bus voltage during Loss of Offsite Power (LOOP). Should the bus voltage be lost the fans will trip and manual action will be required to restart the fans.

Currently, Fans 4V1A, 4V1B, and 4V1C are aligned to the EDGs via the vital section of Motor Control Centers 4B07, B08 and 4B05, respectively, and fan 4V1D is aligned to the EDG via Load Center 4B02. The existing control circuit, automatically restarts all these fans which were running prior to Loss of Offsite Power. This existing feature is being eliminated under this PC/M to reduce loading on the emergency diesel generators.

Safety Evaluation:

The existing design for Normal Containment Cooling Fans 4V1A, 4V1B, 4V1C and 4V1D includes a maintained contact control switch which allows the operating fans to automatically start and load onto the emergency diesel generators in the event of a loss of offsite power. The modifications performed under this PC/M will eliminate the capability of automatic restart upon restoration of bus voltage. Therefore, as required by 10 CFR 50.59, the following evaluation has been performed to determine if this modification constitutes an unreviewed safety question and requires prior NRC approval.

A proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this PC/M for the following reasons:

- The normal containment coolers are non-safety related and no credit is taken for their operation under loss of offsite power conditions to mitigate the effects of accidents considered in Chapter 14 of the FSAR.

- An analysis was perforemd to determine the temperature inside containment without the normal containment coolers operating. This analysis (Reference Bechtel letter SFB-2507, dated May 2, 1986) showed that the temperature would be maintained below 132°F for 72 hours provided that one cooler is started within one hour after the loss of offsite power. This increased temperature over the normal maximum of 120°F was evaluated with respect to safe plant operation and determined to be acceptable for the following reasons:

  - Under accident conditions, i.e., LOCA or MSLB, containment heat removal is accomplished via the emergency containment coolers and containment spray systems. The normal containment coolers are automatically de-energized under these conditions and perform no active heat removal function.

  - An analysis performed by FPL Engineering (Reference JPE-IC-PTP-86-04-001, Revision 0 dated May 1986) determined that containment temperature of 132°F on a non-accident unit would not adversely affect equipment or instruments associated with engineered safeguards provided that the operators can stabilize the plant after reactor trip and maintain the conditions listed below:

    - Pressurizer pressure shall be maintained above 1800 psig through the alternate use of pressurizer heaters an charging pumps. Charging pumps are needed 30 minutes out of every hour, to maintain the unit in hot standby. During the remaining 30 minutes at least 3 banks of pressurizer heaters of 50 kw each should be operated to make up for heat losses during the period without heater operation.

    - RCS pressure should be maintained less than 2258 psig to avoid automatic PORV or safety valve operation.

    - Containment Pressure should be maintained less than 3.0 psig.

    - Steam Generators pressure should be maintained less than 50 psig between SGs.

    - The Steam Generator pressure/steam valve feedback effect prevents the coincidence of SI actuation signals in the SI logic for high steam flow and low SG pressure (600 psig) on low Tavg (531°F). However, operators should cut back on AFW to the SGs to assist in preventing a low Tavg below 531°F.

- The 72 hour time limit specified in the anlaysis was considered appropriate since at that time it is expected that sufficient diesel generator load capability would exist and/or alternate means of containment cooling (i.e., use of purge system) would be possible. In addition the calculation is consistent with the Appendix R safe shutdown analysis which assumes loss of offsite power for 72 hours.

- The operability of safety related electrical equipment and non-safety related electrical equipment whose failure could prevent unsatisfactory accomplishment of safety function is ensured since these components are qualified to much higher temperatures in accordance with the requirements of 10 CFR 50.49.

- All new components and the existing relays used for this modification which are an integral part of, or interface with an existing safety related system are qualified for their intended application and seismically installed, therefore, the integrity and reliability of existing safety related systems will not be degraded.

- All new cables utilized for this modification are qualified for their intended application and have been evaluated for ampacity and voltage drop and determined to be acceptable. In addition, all new cables will be routed in raceway installed seismically thereby precluding any adverse seismic or seismic II/I conditions.

- An evaluation (Reference Bechtel letter SFB-2672 dated May 31, 1986) was performed to address the ability of the normal containment coolers, and their associated discharge dampers to function at elevated temperatures. The results of this evaluation show that operation of the coolers will not be adversely affected at a containment temperature of 132°F. The associated discharge dampers have been walked down and verified on Unit 4 to be suitable for use in an ambient temperature of 130°F. These dameprs are suitable for use since they will open at 1 hour, prior to the containment temperature reaching 130°F.

- As specified in Attachment 5, appropriate procedural controls will be implemented by FPL to ensure that manual operation of the normal containment coolers under loss of offsite power conditions will be performed in a manner such that containment temperatures will not exceed 130°F.

The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created as a result of de-energizing the normal containment coolers under loss of offsite power conditions since no new failure modes are introduced due to the resulting temperature increase. An analysis perforemd by FPL Engineering, as identified above has demonstrated that equipment and instrumentation associated with safeguards actuation will not be affected by the temperature increase. Other safety and non-safety related components required for safe plant operation are qualified for higher temperatures in accordance with the requirements of 10 CFR 50.49.

The margin of safety as defined in the basis for the Technical Specifications is not reduced since neither operation of the normal containment coolers or containment temperature are governed by Technical Specification limits and the normal maximum containment temperature of 120°F would only be exceeded during periods of off-normal operation, i.e., loss of offsite power.

Based on the preceding, this modification does not constitute an unreviewed safety question and prior NRC approval is not required.

NOTE: An additional engineering evaluation is required to demonstrate that an increase in containment temperature above the normal maximum of 120°F will not jeopardize the safe shutdown capability of a non-accident unit under loss of offsite power conditions. The results of this analysis will be incorporated into the safety evaluation as necessary, prior to issuance of the PC/M for review/approval by the PNSC.

PLANT CHANGE/MODIFICATION 86-094

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 07/27/86

SUMMARY DATE: 09/03/86

REVISION: 0

EMERGENCY DIESEL GENERATOR - B ROOM VENT FAN 4V34 POWER SUPPLY

Summary:

This PC/M changed the power and control supply source to the EDG-B Room Vent Fan from MCC 4A Vital section to MCC 4B Vital section. Control Power Transformer (CPT) size is also increased from 50 VA to 100 VA as part of this change. The vent fan operation logic is not changed with this modification. The change in the EDG loading, as a result of this PC/M, is addressed in the PC/M. The separation of raceways for the new cables to meet the Appendix R separation criteria is also addressed in the PC/M. All applicable design verification elements of EDPI 3.16-10, Exhibit H were considered during this review.

Safety Evaluation:

The modifications specified by this PC/M package are necessary to ensure the availability of power and control supply for EDG-B Room Vent Fan when EDG-B operates.

The Vent Fan Motor as a component is not Q-listed and is associated with the EDG-B system which has been identified as Safe Shutdown and Alternate Safe Shutdown systems in Appendix R analysis. The vent fan motor should be added to the FPL Q-list and classified as Important to Safety (ITS) since its operation is important to the operation of EDG-B. The Vent Fan when operating supplements and changes the pattern of air flow inside EDG-B Room. It is, therefore, important that the power and control supply to the Vent Fan Motor should be ensured when EDG-B operates. This PC/M package aligns the Room Vent Motor to EDG-B via MCC 4B vital section and as such ensures the power and control supply to the Vent Fan Motor when EDG-B operates. The present functional control logic of the Vent Fan has not been modified by this PC/M. Except for the relocation of power and control supply, the design maintains the existing automatic/manual control features of the Vent Fan.

The modifications provided by this PC/M are associated with installation of relocated Motor Starter with larger size CPT and wiring in the MCCs. All work associated with MCCs is considered safety related and, therefoe, will be performed under appropriate Quality Assurance Programs. The CPT purchased will match the existing device/equipment quality level.

All cables associated with the modifications are environmentally qualified and all conduit supports will be designed and installed seismically.

Construction activities associated with this PC/M require no unusual techniques or equipment, and will be accomplished in a controlled manner under existing procedures.

None of the work associated with this PC/M is in the Containment Building. All work associated with this PC/M is in low radiation areas and, therefore, ALARA criteria is not applicable.

Penetrations through fire barriers and/or associated barrier seals, wherever needed, will be sealed with three-hour rated seals to maintain the integrity of the fire barriers in accordance with the design documents. Conduits penetrating a fire barrier are also sealed internally, as required, in accordance with Drawing 5610-A-182.

No work associated with this PC/M changes the ECCS heat sink analysis, nor affects radioactive waste treatment, safety related snubbers, spent fuel pit cooling systems, or effluent monitoring system.

Cable routing for this PC/M has been reviewed in accordance with the requirements of 10CFR50, Appendix R, and found acceptable.

Equipment or cable associated with this PC/M will not be attached to or installed in the proximity of any block walls which have not been previously analyzed to preclude their failure and subsequent damage to adjacent safety related equipment.

This modification has been reviewed with respect to the NRC Safety Evaluation Report, "Susceptibility of Safety Related System to Flooding from Failure of Non-Category 1 Systems", dated September 4, 1979, and is acceptable for the following reasons:

- Identified modifications will not create new more limiting sources of flooding, or adversely affect the design or operation of any flood mitigating features.

- Wiring added by this PC/M are located in existing equipment and as such, this PC/M does not add new equipment which may be suceptible to flooding.

As required by 10 CFR 50.59, the following evaluation has been performed to determine if this modification constitutes an unreviewed safety question and requires prior NRC approval. A proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this PC/M for the following reasons:

- The control logic of the vent fan has not been changed.
- The relocation of the power and control supply enhances the availability of the vent fan when EDG-B is operating.
- All new components utilized for this modification are qualified for their intended application. Cables have been evaluated for ampacity and voltage drop and found acceptable. All new cables will be routed in raceway installed seismically thereby precluding any adverse seismic or seismic II/I conditions.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created since:

- Diesel generator loading has been evaluated for this modification and found acceptable.
- The functional control logic for the vent fan has not been modified.
- This modification does not change the performance capability of the vent fan.

The margin of safety as defined in the basis for the Technical Specifications is not reduced since the operation of the vent fan is not governed by Technical Specification limits.

Based on the preceeding this modification does not constitute an unreviewed safety question and prior NRC approval is not required.

PLANT CHANGE/MODIFICATION 86-095 PC/M CLASSIFICATION: NS

UNIT 3

TURNOVER DATE: 07-24-86

SUMMARY DATE: 09-03-86

REVISION: 0

PROVIDE NEW POWER FEED FOR NON-VITAL SECTION OF MCC 3A AND D

Summary:

The intent of the design implementation package is to resolve a concern in the existing design that a single failure of the tie breaker between vital and non-vital busses of Motor Control Centers (MCC) 3A and D can disable both Channel A and B diesels in providing emergency power.

This design modification provided a redundant means of isolating non-vital loads from the vital power source on a loss of offsite power (LOOP) or on a LOOP together with Safety Injection Actuation Signal (SI).

This was achieved by physically separating the non-vital bus from the vital bus at MCCs 3A and D and feeding the non-vital Busses directly from the same safety related Load Centers 3A and D which provides the normal source of power to the vital busses of MCCs 3A and D. The feeder breaker at the load center and the tie breaker converted to the incoming breaker at the MCCs will be in series, thereby providing the necessary redunancy for the single breaker failure to trip.

Since the trip logic for the tie breaker (now the incoming breaker to non-vital bus) has not been changed the previously reviewed design philosophy is not affected. On the other hand, the original design is improved by tripping the MCC feeder breaker at the load centers without any intentional time delay for the design basis events mentioned above, thus quickly isolating the non-vital loads from the vital source of power.

The raceways and cables are installed by the PC/M 86-093. The installation meets all the criteria required for safety related equipment installation, including seismic, Appendix R, blockwall, flooding, and nuclear qualification requirements. The new breakers added at Load Centers 3A and 3D are existing nuclear qualified spares. The modification to the bus work at MCCs 3A and D will not violate the original safety qualification for the equipment. This is based on confirmation by the equipment vendor who will provide documentation to substantiate this determination.

The design verification has considered the elements presented in exhibit H of the EDPI 3.16-10 and a check has been made to verify compliance to the applicable Project Standards and licensing commitments.

Safety Evaluation:

Per the existing plant design, the non-vital section of MCC 3A (D) is fed from the vital section of MCC 3A (D) through a tie breaker, and is automatically shed after a time delay on a loss of offsite power (see Section 2.0) by tripping of the tie breaker. By implementation of this PC/M, the power feed for MCC 3A (D) non-vital bus is relocated form the vital bus of MCC 3A (D) to the source Load Center 3A (3D), and the existing tie breakers will be used as the incoming breaker at the MCC non-vital bus NV3A (NVD). NV3A (NVD) will be shed on loss of load center bus voltage, with no intentional time delay, by independent trip circuits to the new load center Feeder Breaker 30103 (30411) and the MCC incoming Feeder Breaker (30535 (0832). Therefore, as required by 10CFR50.59, the following evaluation has been performed to determine if this modification constitutes an unreviewed safety question and requires prior NRC approval.

In accordance with 10CFR50.59 a proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be, increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; of (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

To evaluate the above three conditions, every change resulting from this PC/M, as categorized below, has been investigated.

- Electrical Distribution System Operating Characteristics
- System Response to LOOP/SI
- Re-energization of Non-Vital MCC NV3A (NVD) Loads During Plant Recovery
- Appendix R Compliance
- Effects of New Component Installation

PLANT CHANGE/MODIFICATION 86-041 PC/M CLASSIFICATION: SR

UNIT: 3/4

TURNED OVER DATE: 7/25/86

SUMMARY DATE: 10/20/86

REVISION: 0

Modification of MCC "D" Automatic Transfer

Summary:

PC/M 86-041 provides for modification of the Automatic Transfer Circuit of MCC "D". This evaluates single failures due to (1) Failure of Emergency Diesel Generator "B", (2) Failure of Battery 3B/Sequencer 3B, or (3) Failure of Breaker 3AB20. The circuit design prior to PC/M 86-041 modifications did not ensure an automatic transfer in case of failure of Battery 3B/Sequencer 3B or failure of Breaker 3AB20 for certain conditions. This change request provides circuit modifications for the above failures for various plant conditions. Automatic transfer of MCC "D" on loss of offsite power with concurrent safety injection actuation signal on both Units is also addressed.

Safety Evaluation:

- This modification eliminates single failure susceptibilities in the MCC "D" automatic transfer scheme associated with scenarios listed in Table 1. The probability of occurrence of an accident previously evaluated in the FSAR is not increased because the devices being added are environmentally and seismically qualified and seismically installed, effects on the existing cabinets have been evaluated, and no other safety related system functions or operations are jeopardized.

  - The devices and cable added by this PC/M are nuclear qualified. This modification does not alter the operation or function of any other plant safety system. By eliminating single failure susceptabilities in the MCC "D" automatic transfer circuit, the availability of the MCC is increased. The existing transfer logic is retained. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

  - The relays in the parallel logic added by this PC/M are applied in a fail-safe configuration. Both relays on two load centers must drop out and remain dropped out for 40 seconds to initiate an MCC "D" transfer. The existing permissives must also be satisfied to effect the transfer. Therefore, the probability of occurance of an equipment malfunction important to safety previously evaluated in the FSAR is not increased by this modification.

PC/M 86-041 Modification of MCC "D" Automatic Transfer

° This modification improves the reliability of the MCC "D" transfer scheme and the availability of MCC "D" and the emergency containment coolers by eliminating single failure susceptibilities. Should Battery 3B fail in conjunction with a loss of offsite power on Unit 3 or Breaker 3AB20 fails to close without safety injection signal present, implementation of this modification ensures that power will be available to MCC "D". This modification does not alter the operation or function of any other safety related systems. The EDG loading is not changed for the single EDG case where one EDG has failed. However, for the two EDG cases where a single failure other than loss of an EDG has occured certain loads will not be stripped when MCC "D" transfers to its alternate supply. Since the auto loading of the EDG is within the Technical Specification limitation, the consequences of an equipment malfunction important to safety previously evaluated in the FSAR is not increased by this modification.

° The relays in the parallel logic added by this PC/M are applied in a fail-safe configuration. Both relays on two load centers must drop out and remain dropped out for 40 seconds to initiate and MCC "D" transfer. There are no interfaces with non-safety systems. The devices added are seismically and environmentally qualified and seismically installed and the effects on the existing cabinets have been evaluated. The cable is environmentally qualified and seismically installed. The loading on the emergency diesel generators is not increased. This modification does not alter the operation or function of any other safety related system. The existing MCC "D" transfer logic is retained, and this modification does not alter the operation or function of any other safety related system. MCC "D" and emergency containment cooler availability is improved. Therefore, the possibility of an accident of a different type than any previously evaluated in the FSAR is not created.

° This modification does not reduce the integrity, operation or function of any other safety related system addressed in the Technical Specifications. This modification improves the availability of MCC "D". Therefore, the margin of safety as defined in the basis for any plant Technical Specification is not decreased.

Based on the preceding, this modification to the MCC "D" automatic transfer scheme does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 78-102B

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 07-17-86

SUMMARY DATE: 11-06-86

REVISION: 1

STEAM GENERATOR BLOWDOWN RECOVERY SYSTEM

Summary:

The Steam Generator Blowdown Recovery System was designed to maintain the required chemistry of the steam generator and for maximum water and heat recovery rate of 1% blowdown of maximum feedwater flow. Blowdown in excess of 1% was designed to be discharged into the discharge canal.

Safety Evaluation:

The probability of occurrence or the consequences of an accident or malfunction of equipment important to the safety of the plant, previously evaluated in the FSAR, has not been increased. There is no possibility of an accident or malfunction different than those previously evaluated. Therefore, it can be concluded that this PC/M does not pose any unreviewed safety questions.

PLANT CHANGE/MODIFICATION 84-02

PC/M CLASSIFICATION: NS
UNIT: 4
TURNED OVER DATE: 09/15/86
SUMMARY DATE: 11/12/86
REVISION: 0

MODIFICATION TO COMPLY WITH R.G. 1.97, REV. 3 REQUIREMENTS TO PROVIDE QUALIFIED LIMIT SWITCHES

Summary:

This modification consisted of replacing existing safety related non-qualified limit switches for the Reactor Drain Tank (RCDT) and Component Cooling Water (CCW) containment isolation valves with fully qualified and documented Namco Series EA 180 limit switches.

Safety Evaluation:

This modification does not degrade the system or equipment as follows:

1. The qualified limit switches will extend the environmental and electrical integrity of the existing switches.

   a. No system characteristics will be changed and the probability of occurrence of an accident would be no greater.

   b. The consequences of an accident previously analyzed in Chapter 14.0 would not be altered.

   c. There is no potential to jeopardize the operation of other safety related systems.

   d. The consequences of equipment malfunctions are no more severe than previously evaluated in FSAR Chapter 14.

2. The qualified limit switches do not decrease the design margins of the system, change the operation function of conditions, or affect other safety related equipment.

   a. This change would not create the possibility of an accident not considered in FSAR Chapter 14.

   b. The replacement limit switches would not create the possibility of malfunction of equipment not considered in Chapter 14 of the FSAR.

   c. This modification will not decrease any margin of safety discussed in any technical specification.

PLANT CHANGE/MODIFICATION 83-63

PC/M CLASSIFICATION: NNSR

UNIT: 3

TURNED OVER DATE: 12/12/85

SUMMARY DATE: 2/9/87

REVISION: 0

Improved Floor Drain for the Containment Spray Pump Room

Summary: This change modified the floor drain in the Containment Spray Pump Room to allow for better drainage. The drain piping was rerouted to provide a more direct flow path to the drain header.

Safety Evaluation: This modification improved the containment spray pump room to function as intended. The change was not nuclear safety - related. This modification did not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, did not create the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR and did not reduce the margin of safety as defined in the basis for any technical specification.

F2

PLANT CHANGE/MODIFICATION 83-114

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 4/4/86

SUMMARY DATE: 2/16/87

REVISION: 0

Unit 4 Reactor Cavity Filters - Lead Shielding

Summary: Lead shielding and its associated supports were added inside the containment structure at the 58 foot elevation. The added shielding decreases the radiation dose rates in the surrounding area.

Safety Evaluation: The shielding system does not perform a nuclear safety related function. Since it is installed in proximity to safety injection - piping, it is built to withstand the maximum earthquake loading used for the design of Turkey Points Units 3 and 4 seismic category I structures in accordance with the FSAR. This modification does not increase the possibility of occurence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR and does not reduce the margin at safety as defined in the basis for any technical specification.

F2

PLANT CHANGE/MODIFICATION 86-003

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 8/07/86

SUMMARY DATE: 2/20/87

REVISION:

MSIV NITROGEN SUPPLY ADDITION FOR UNIT 4 (INTERIM)

Summary:

The PC/M provided an interim system to supply nitrogen to the 3 Main Steam Isolation Valves. This system manually actuated and serves as a backup to the existing Instrument Air Supply, to ensure that the MSIV's could be maintained closed in the event of a small steam leak downstream of the MSIV's, and the Instrument Air System is inoperable.

This system is designed in accordance with respect to FSAR paramenters and NRC concerns on valve closure.

The interim system does not, however, address the specific requirements to achieve a second MSIV closure, single failure and other operability concerns; these will be addressed in the final (and permanent) Nitrogen Backup System Design.

Safety Evaluation:

This temporary change is the first step to upgrade the MSIV's to meet the FSAR closure requirements of five second closure with no steam flow and loss of instrument air. The operating limits on the instrument air system have been specified, which permits power operation of the plant. The change does not affect the operability of the Instrument Air System with respect to closing the MSIV'S.

It is noted that the Nitrogen Supply System operability must be verified when the system is placed in service (i.e. when air pressure < 66 psig and MSIV will not close) in order to ensure that the MSIV's can be closed in the event of a postulated accident.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-064 (PWO)

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 8/07/86

SUMMARY DATE: 2/23/87

REVISION (PWO): 0

TITLE: 4.16K V FUSE HOLDER SUBSTITUTION

Summary:

This CPWO provided for replacement 4.16 KV SWGR fuse holders.

Safety Evaluation:

A Safety Evaluation has been performed which approves the use of the replacement fuse holders. The evaluation resulted in the following determination.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

F2

PLANT CHANGE/MODIFICATION 86-70 PC/M CLASSIFICATION: NNSR

UNIT: 4

TURNED OVER DATE: 6/24/86

SUMMARY DATE: 2/26/87

PC/M REVISION: 0

TITLE: REPAIR DAMAGED T/C CABLES AT TE-4-1418 & TE-4-1421

Summary: Ables 4T-2000/DDPS-TE-4-1418/1 for temperature element TE-4-1418 and repairable at the condulet as documented by NCR 109-86.

This modification consisted of installing a terminal box in the existing conduit run 4A422 and providing new thermocouple cables from the terminal box to temperature elements TE-4-1418 and TE-4-1421. This terminal box and new cables will replace the existing damaged cables as documented by NCR the condensate pump suction header piping.

Safety Evaluation:

This modification includes the installation of anew terminal box (TB4927) and routing new thermocouple cables from the terminal box to TE-4-1418 and TE-4-1421 in order to replace the existing damaged cables. The entire modification is non-safety related consistent with the existing installation.

This modification is non nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

F2

PLANT CHANGE/MODIFICATION 86-71 PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 6/24/86

SUMMARY DATE: 2/20/87

REVISION:

ICW BASKET STRAINER REPLACEMENT

Summary:

This CPWO requested the replacement of the 4A ICW/CCW Basket Strainer due to the extensive corosion of the existing strainer. The new strainer was fabricated to ASME Sect VIII Div.1 1983ED-1985 Summer Add.; the existing strainer was built to ANSI B31.1 requirements. The original strainer was built by Zurn Industries, and the new strainer was built by Zurn's new owner Hayward Industrial Products to the same dimensional specification, and essentially the same strainer body material. The new strainer was coated with Belzona E-C Barrier ceramic material, while the original strainer was coated with coal-tar epoxy.

Safety Evaluation:

All Dimensions and materials were provided on a one-to-one basis for the new strainer, and the new coatingis considered to be an improvement. The maximum specified nozzle loads for the strainer were found to be acceptable.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 82-83 PC/M CLASSIFICATION: NNS/QA/QC

UNIT: 3 & 4

TURNED OVER DATE: 7/24/85

SUMMARY DATE: 2/24/87

PC/M REVISION: 0

TITLE: ADD BACK-UP TO RELAYING IN SWYD.

Summary:

The PC/M provided the design for installation of a solid state secondary relay system for station and Davis 240 KV lines protection, and the installation of additional breaker failure protection for the 240 KV generator breakers.

Safety Evaluation:

The PC/M is non-nuclear safety related as the additional relaying does not perfrom a safety related function and is so located that it cannot affect any safety related equipment.

No unreviewed safety question exists since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-044

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 12/01/86

SUMMARY DATE: 2/20/87

REVISION: 0

UNIT 4 ICW BASKET STRAINER BELZONA LINING

Summary:

The original strainers were coated with coal tar epoxy, which degraded significantly over the years, which contributed to the extensive corrosion of the strainer body. The new strainer is coated with Belzona E-C barrier ceramic material, and should provide greater protection due to it's superior adhesive strength, bonding strength, tensile strength and toughness. The Belzona lining has demonstrated these characteristics elsewhere in the ICW system and other saltwater service applications.

Safety Evaluation:

The substitution of coal tar epoxy with Belzona provides greater protection, and the probability of the Belzona material disbonding from the strainer is no greater than that of the coal tar epoxy.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-087

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 12/31/86

SUMMARY DATE: 2/20/87

REVISION: 0

CABLE REPLACEMENT FOR MSIV SOLENOID FOR UNIT 4

Summary:

This PC/M provided the replacement of damaged control cables to the solenoids for Main Steam Isolation valves POV-4-2604 and POV-4-2605, (for damage description see NCR's #799-86 & 800-86). The original cables are no longer manufactured, so this PC/M provides a replacement cable with the same characteristics and Dwg. size as the original cable. The new cable is installed in a seismically supported conduit (cable #C24) and connect cable TB4028 to the MSIV solenoids for the aforementioned valves.

Safety Evaluation:

The replacement cable is of the same size and serve the same loads, are qualified for the working enviornment and are seismically supported. No interfacing system loads are affected by the new cables.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 85-152

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 12/18/86

SUMMARY DATE: 02/24/87

REVISION:

NIS INPUT TO TURBINE RUNBACK - REINSTATEMENT OF 1/4 CONFIGURATION

Summary:

PC/M 85-103 provided an interim solution to the problem of NIS caused runbacks by changing the Unit 4 logic for initiating a turbine runback caused by a negative flux rate input (NIS signal) from a 1/4 to a 2/4 configuration. The PC/M stated that the "permanent solution will be forthcoming through implementation of PCM 84-211 during the next Unit 4 scheduled outage." Because PC/M 84-211 assumes the previous 1/4 configuration, a new PCM, 85-152, is required to change the affected portion of the system back to a status identical to the one before implementation of PC/M 85-103.

The task of this design package is therefore, to reinstate the system to its' original 1/4 logic for NIS initiated turbine runbacks, as a necessary precondition to implementation of PC/M 84-211.

Safety Evaluation:

This charge does not constitute an unreviewed safety question for the following:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not changed relative to this change. The logic for NIS initiated runbacks is changed back to its original 1/4 configuration. Therefore, the change would not prevent a runback from occuring to a single dropped rod.

2. The possibility for an accident or malfunction of a different type than has been proviously evaluated in the FSAR is not created due to the re-instatement of the logic to its original 1/4 state.

3. The margin of safety as defined in the bases for the technical specifications has not been decreased.

   Based on the above discussion, we have concluded that there is no unreviewed safety issue associated with this modification. There is no change in the technical specifications. Therefore, the proposed change in the turnbine runback logic on NIS signal has been demonstrated to be acceptable.

F2:

PLANT CHANGE/MODIFICATION 85-085

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 09/30/86

SUMMARY DATE: 03/03/87

REVISION: 0

AUXILIARY FEEDWATER CV INSTRUMENT AIR FILTRATION MODIFICATION

Summary:

This modification provided for the installation of new filters in the instrument air (nitrogen backed) supply line to each unit 3 Auxiliary Feedwater Control Valve as shown on the drawing listed on Attachment 1. The installation of the filters provides better quality air to the Control Valve positioners and actuators.

The new filters were installed upstream of the instrument air-nitrogen supply tie-in connections in the instrument air supply line. The filters are provided with isolation valves and bypass lines with a valve for ease of maintenance without isolating the Control Valve air supply. A new anchor, and supports as required per 5177-PS-21 were installed to isolate the filter assembly from the downstream check valve. This isolation was required to ensure the functional and structural integrity of the safety related portion of the system. The new piping is 1/2-inch galvanized steel, Schedule 40.

The new filters are Parker Hannifin Standard Airline Filters designed to separate dirt particles and water. Sizing for this installation was determined by the maximum flow capacity required for operation of the positioners on the Auxiliary Feedwater Control Valves. Using the flow rates required with 100 psi instrument air and 60 psi instrument air as the pressure range to operate the positioner, (13 scfm and 8.5 scfm respectively), and a required 2 to 5 psi pressure drop across the filter for efficient operation, and the particle size filtration needed. in this case 5 micron, a filter size selection was made from a chart supplied by the filter manufacturer. Therefore, the filter size selected is specific for this application, and has the concurrence of the filter manufacturer. This analysis is documented in Calculation M08-0424-01.

The method of operation of the air filter is such that pressurized air (instrument air at 100 psig) flows through a louvred deflector and is directed into a swirling pattern. Liquids and large dirt particles are thrown against the inside wall of the see-through polycarbonate bowl by cyclonic action and fall into a quiet zone below the lower baffle. This lower baffle maintains the quiet zone to prevent turbulent air from returning liquids and solids into the air stream. The instrument air, now free of liquids and dirt particles, passes through a filter element sized to remove particles down to 5 microns.Clean air then flows through the outlet port. Liquids are discharged from the bowl by the automatic drain valve. The Parker Hannifin model number for this installation is 04F15B which translates to the following:

04F - Minature Series
1 - 1/4" NPTF Port Size
5 - Automatic Drain
B - 5 Micron Filtering Element

Safety Evaluation:

These modifications provide for the installation of new filters in the instrument air supply line to each Unit 4 Auxiliary Feedwater Flow Control Valve. The air filters, their associated isolation and bypass valves and piping were not installed to the requirements of seismic Category I. The seismic boundary anchor assures the structural integrity of the piping and components downstream of the anchor. The modifications provided by the PC/M include passive components whose function will not be impaired by any design basis accident described in the FSAR.

These modifications did not introduce new safety related equipment which could be affected by fire or add new combustibles which could invalidate the Fire Zone Heat Loading Analysis previously submitted to FPL per Bechtel letter SFB-1741 dated April 24, 1985. In addition, these modifications did not adversely affect any existing or proposed fire protection features of the plant. Therefore, these modifications do not affect the Turkey Point Fire Protection Program.

These modifications are not inside containment, are not attached to block walls, do not involve safety related snubbers and do not impact spent fuel pool cooling operations of the plant.

No special ALARA considerations were required because the modifications were carried out in the areas outside of Radiation Control Area.

The modification did not involve the addition of electrical cable or any changes to existing raceways. The final modifications accomplished by this PC/M did not affect the flooding analysis as described in the NRC Safety Evaluation Report dated September 4, 1979, because they do not introduce a new source of flooding, modify the existing flood mitigating features, or install or modify any safety related components which could be affected by flooding.

Based on the preceeding, the following conclusions can be made:

- The probability of occurrence of an accident previously evaluated in the FSAR was not increased because the modifications do not alter the function of Instrument Air or Nitrogen Backup Supply to AFW Control Valves.

- The consequences of an accident previously evaluated in the FSAR was not increased because the added tubing, filters and valves do not affect the performance of safety features.

- The changes which are associated with plant safety features are minor in nature and do not change the function of the plant safety features. Therefore, there is no possibility that an accident may be created that is different than any already evaluated in the FSAR.

- The probability of occurrence of an equipment malfunction important to safety which had already been evaluated in the FSAR, will not be increased due to the installation of filters and isolation valves because this change did not adversely affect any equipment important to safety.

- The consequences of equipment malfunction important to safety which had already been evaluated in the FSAR was not increased because the performance design basis was not changed from that described in the FSAR.

- The modification to the AFW Control Valve Instrument Air Supply System identified in this PC/M did not change the inherent function or design basis for the system. In addition, the in-line air filters will be inspected and cleaned (if necessary) on a regular basis in accordance with approved system maintenance procedures, to prevent possible blockage. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than previously evaluated in the FSAR has not be created.

- This modification provides cleaner air to the AFW Control Valve actuator and thus does not reduce the margin of safety as defined in the bases for any Technical Specification.

Based on the above, these modifications do not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-139

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3 & 4

TURNED OVER DATE: 08/20/86

SUMMARY DATE: 03/05/87

REVISION: 0

HALON SUPPRESSION SYSTEM FOR APPENDIX R MODIFICATIONS

Summary:

The halon suppression systems are provided in response to licensing commitments to satisfy 10 CFR 50 Appendix R, Section III.G requirements as described in the FPL Fire Protection Review report.

This modification provides for the installation of halon suppression systems in the Cable spreading Room (Fire Zone 98) and the Inverter Rooms (Fire Zones 108A and 108B). The halon suppression systems provided are automatic, total flooding type utilizing Halon 1301. The systems are designed to provide a concentration of 6.0 to 6.5 percent, by volume, within 10 seconds of actuation, and to maintain that concentration for a minimuim of 30 minutes. The concentration is based on industry standards and consideration for potential hazards to personnel.

The system consists of cross-zoned ionization detectors for actuation, local control panel, piping distribution system, and halon main and reserve supply for each area protected.

Safety Evaluation:

The probability of occurrence of an accident previously evaluated in the FSAR is not increased because these modifications do not change the function or arrangement of safety related features.

The halon suppression system is installed to Seismic II/I requirements and the discharge of halon will not thermally affect any sensitive electrical equipment. The halon storage cylinders do not need a missile shield above them because the bottles are only pressurized to 360 psi, the bottle mounted control heads are connected to a substantial flexible hose, and the framing and header piping arrangement would impede control head travel.

With respect to the consequences of an accident previously evaluated in the FSAR:

All modifications are seismically installed. Therefore, the modification does not affect the consequences of any accident previously evaluated. In fact, the halon suppression system will mitigate the consequences of a fire in the area.

With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR:

The modification is not associated with equipment important to safety previously evaluated in the FSAR and the modifications will not impact safety related equipment since the modification additions are seismically supported and thermally will not impact sensitive electrical equipment. The halon suppression system in fact adds a margin of safety by reducing the threat of exposure fires on equipment important to safety.

With respect to the consequences of malfunction of equipment important to safety previously evaluated in the FSAR:

The consequesces of equipment malfunction important to safety which have already been evaluated in the FSAR is not increased because the suppression system is seismically supported.

With respect to the possibility of an accident of a differenc type than any analyzed in the FSAR"

There is no possibility that an accident or malfunction of equipment important to safety may be created which is of a different type than any already evaluated in the FSAR because these modifications are not associated with safety related systems and are seismically supported.

With respect to the possibility of a malfunction of a different type than any analyzed in the FSAR:

The modifications installing the halon suppression systems do not provide any interaction with any safety related equipment and the systems are seismically installed. Therefore, the possibility of a malfunction of a different type than any analyzed in Chapter 14 of the FSAR is not created.

With respect to the margin of safety as defined in the basis for any Technical Specification:

These modifications relate to the Technical Specifications dealing with fire protection. The marg' s of safety, as defined in the associated bases, are not reduced because these modifications do not prevent the safety features from performing their intended safety functions. In fact these modifications tend to increase the margin of safety by decreasing the probability that performance of safety related features will be hindered by fire.

Based on the preceding these modifications do not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-56

PC/M CLASSIFICATION: NNSR

UNIT: 3 & 4

TURNED OVER DATE: 10/31/86

SUMMARY DATE: 2/27/87

REVISION: 0

REPLACEMENT OF INTAKE WALKWAY

Summary:

This PC/M replaced the Intake Access Walkway at the south retaining wall of the Intake Canal. The walkway was replaced with epoxy coated pre stressed slabs which were a one for one replacement of the old slabs. The same anchoring methods were used for the new walkway with the exception of the eastern-most slab. The anchorage was approved per FCN #607. The existing handrail was replaced with a minor change to nounting to make it removable (FCN #509).

Safety Evaluation:

The Intake Access Walkway is non-nuclear safety related and was replaced with one for one replacement walkway. Sufficient care was taken during the construction of this walkway to ensure the south retaining wall on the intake canal was not altered from its existing configuration. This change did not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, did not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR and did not reduce the margin of safety as defined in the basis for any technical specification.

PLANT CHANGE/MODIFICATION 84-210

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 08-19-86

SUMMARY DATE: 09-09-86

REVISION: 0

TURB. E RUNBACK MODIFICATIONS

Summary:

This modification increases the availability/operability of the plant by enabling operations to remove an unreliable input from the turbine runback logic and still permit the use of automatic rod control. The following modifications were performed:

Reconnected the bank selector switch auto contacts which were disconnected by PC/M 83-88.

Multiplied the Rod-on-Bottom signals in the Rod Position Indication (RPI) rack to provide two separate RPI signals into the turbine runback initiating logic.

Modified the governor runback and load limit runback logics so that either an RPI or a one-out-of-four Nuclear Instrumentation System (NIS) flux rate signal (when selected) will initiate both the turbine governor and load limit runbacks.

Installed a four position, key-locked turbine runback selector switch on the control console.

Disconnected the contacts on the defeat switch for the RPI input to the turbine runback logic and the control room annunciator.

Combined annunciator windows B1-7 with B2-7 and located the new alarm on window B1-7, and

Modified the load limit runback logic so that a steam generator feedwater pump breaker trip with turbine first stage pressure above 60 percent load will automatically initiate a turbine runback.

Provided an alarm via the annunciator system to indicate when the new selector switch is out of the normal (RPI) position or when the logic matrix for the RPI portion of the selector switch fails to actuate.

Safety Evaluation:

Some of the primary circuits that provide signals to Turbine Runback Logic are Nuclear Safety Related. However, actual circuitry that initiates the runback logic is not safety related. There is no unreviewed safety question since NIS/RPI signal selection for turbine runback logic initiation was part of original design and no devices installed by this PC/M penetrate pressure boundary or affect any piping system analyses. None of the equipment will be installed adjacent to any block wall, and no equipment by this PC/M shall be installed inside the containment. It does not involve a significant increase in the probability or consequences of an accident previously considered and does not involve a significant decrease in safety margin.

PLANT CHANGE/MODIFICATION 85-124

PC/M CLASSIFICATION: NNS

UNIT: 3/4

TURNED OVER DATE: 10/31/86

SUMMARY DATE: 02/26/87

REVISION: 0

REMOVAL OF SPENT RESIN PIPE IN LAUNDRY ROOM

Summary:

This modification removed unused portions of the resin transfer lines. Modification cut and capped thoes portions of the piping system that remained in the auxiliary building and physically removed thoes portions no longer needed. The benefits of the modification included lower dose rates in the laundry room.

Safety Evaluation:

This modification is non-nuclear safety related with no unreviewed safety question since the probability/consequences of an accident proviously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

F2

PLANT CHANGE/MODIFICATION 86-22

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 9-8-86

SUMMARY DATE: 4-13-87

REVISION: 0

ICW HEADER ISOLATION VALVE REPLACEMENT:

Summary:

This PC/M replaced ICW. Isolation valves 4-50-307 through 4-50-310 with valves of like kind valve 4-50-307 exhibited gear leakage. Internal inspection found that valve seat of valves 4-50-308 thru 4-50-310 had lost their resiliency and may also be leaking.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 84-124

PC/M CLASSIFICATION: SR
UNIT: 4
TURNED OVER DATE: 3/9/87
SUMMARY DATE: 4/8/87
REVISION: 0

UNIT 4 AUXILIARY FEEDWATER FLOW TRANSMITTER REPLACEMENT

Summary:

This PC/M replaced Unit 4's Barton ITT-752 AFW flow transmitters with Rosemount 1153 flow transmitters. Transmitters replaced were FT-4-1401A & B, FT-4-1457A & B and FT-4-1458A & B.

Safety Evaluation:

This PC/M is safety related; the new transmitters are fully qualified. The probability of occurrence, or the consequences of a design basis accident, or malfunction of equipment important to safety, as previously evaluated in the FSAR, will not be increased.

PLANT CHANGE/MODIFICATION 84-158

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 12-17-85

SUMMARY DATE: 04-13-87

REVISION: 0

EDG "4B" CFD RELAY REPLACEMENT

Summary:

This PC/M improves the fragility level of the D/G differential circuit by reducing the probability of relay trip due to mechanical vibration. This modification is accomplished solely by replacing the existing differential relays and cases, while implementing no internal or external wiring changes in the diesel generator control panel. This then precludes any new type of interaction with other safety related equipment. Therefore, this PC/M is nuclear safety related but does not involve an unreviewed safety question.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 83-141

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 5-21-86

SUMMARY DATE: 04-13-87

REVISION: 0

FIRE BARRIERS FOR APPENDIX R MODIFICATIONS

Summary:

The scope of this PCM covers the installation and upgrading of Unit 3 fire barriers in accordance with the requirements of Section III.G of Appendix R to 10CFR50. The purpose of these fire barriers is to control or restrict the spread of fire from one fire area to another as identified in the Appendix R fire Protection Review Report. The type of modifications will include stairwell enclosures, addition of new fire barriers, upgrading of existing fire barriers and addition of curbs around electrical manholes.

Safety Evaluation:

1. The probability of occurrence of an accident previously evaluated in the FSAR is not increased because the fire barriers, and portions thereof, are designed for all applicable loads, including seismic, and the new barriers will not affect the function or operating conditions of safety related equipment.

2. A fire is not postulated to occur simultaneously with a design basis accident. However, since the barriers limit the threat of simultaneous fire exposure to systems, or portions thereof, which are redundant in the performance of safe shutdown functions, the consequences of an accident have not been increased.

3. There is no possibility that an accident will be created which is of a different type than any already evaluated in the FSAR because the only effect of the fire barrier installations is to enhance previously established fire area boundaries and does not diminish the quality of inter-spacial relationships of these areas for which credit may be taken for HVAC. In addition, the barriers have been designed seismically to preclude any interaction with safety related equipment.

4. The probability of occurrence of equipment malfunctions important to safety which have already been evaluated in the FSAR will not be increased because the intent of Appendix R and the design of these barriers precludes a common-threat exposure fire to safety features required for safe shutdown. The modifications under this PCM have no affect on the equipment important to safety.

5. The consequences of an equipment malfunction important to safety which has already been evaluated in the FSAR are not altered by the addition of these fire barriers. Furthermore, since safe shutdown equipment in one area is protected from an exposure fire in another area where operabitity of the redundant safety function equipment may be compromised, the consequences of equipment malfunction has not been increased.

6. There is no possibility that a malfunction of equipment important to safety may be crerated which is of a different type than any already evaluated in the FSAR because the threat of exposure fires, component-for-component, remains unchanged by this modification.

7. This modification relates to Technical Specification 4.15 with regard to fire barriers. This change does not reduce the margin of safety as defined in the associated bases for limiting conditions for operation.

Based on the preceding, this modification does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-145

PC/M CLASSIFICATION: NS

UNIT: 3 & 4

TURNED OVER DATE: 8-14-86

SUMMARY DATE: 4/2/87

REVISION: 0

FIRE DAMPERS FOR APPENDIX R MODIFICATIONS

Summary:

These modifications provide for the installation of seismically qualified fire dampers in the fire barrier penetrations of the Auxiliary Building ventilation system and the Control Building ventilation system as shown on Drawings 5610-M-85/83-145 and 5610-M-91/83-145. The installation of these fire dampers is required to ensure the integrity of three hour fire rated barriers separating redundant trains of safe shutdown equipment, and is necessary to meet Appendix R fire protection commitments defined in the Turkey Point Fire Protection Review Report. These modifications also provide for the installation of isolation dampers used for the halon suppression system in the Control building ventilation system ductwork. The halon suppression system, along with wiring assoicated with damper actuation, is provided for under PCM 83-139. These modifications also provide for the necessary non-safety related ductwork modifications required for installation of the fire and isolation dampers.

Safety Evaluation:

The porbability of occurrence of an accident previously evaluated in the FSAR is not increased because these changes do not alter the design basis of the facility with respect to any system or component required for plant safety.

The consequences of an accident previously evaluated in the FSAR will not be increased because

- The affected conduit, air lines, and piping of systems and components which are safety related that have to be rerouted due to the damper installation have been evaluated. These changes are minor and will not affect the design basis or function of the systems.

- The affected system and components which are not required for safe shutdown would not affect the performance of safety features during installation and as a result of operation or misoperation of the fire dampers. In addition, the change does not alter the basis of the ventilation systems operation as defined in the FSAR

There is no possibility that an accident will be created which is of a different type than any already evaluated in the FSAR because

- Modifications to safety related systems will only be accomplished in accordance with applicable Technical Specification requirements and when plant conditions are as specified in the PCM, and

- The changes which are not associated with any plant safety features do not create any conditions that could be associated with or be more limiting than any accidents defined in the FSAR.

Therefore, there is no possibility that an accident may be created that is of a different type than any already evaluated in the FSAR.

The probability of occurrence of equipment malfunctions important to safety which have already been evaluated in the FSAR will not be increased as a result of the installation of fire dampers. Additional assurance is provided for fire damper modifications being implemented under qualified Quality Assurance and Quality Control Programs.

The consequences of an equipment malfunction important to safety which have already been evaluated in the FSAR will not be increased because the performance design basis has not been changed from that described in the FSAR.

The modifications to the ventilation system and any other existing system identified in this PCM will not change the inherent function or design basis for the systems. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any already evaluated in the FSAR, will not be created.

Portions of these modifications which are associated with safety related or fire protection systems shall be accomplished in accordance with Technical Specification requirements. Therefore, there is no reduction in the margin of safety as defined in the bases for any Technical Specification.

PLANT CHANGE/MODIFICATION 84-2? PC/M CLASSIFICATION: NNS-QAQC

UNIT: 3 & 4

TURNED OVER DATE: 04-25-86

SUMMARY DATE: 04-13-87

REVISION: 0

POST ACCIDENT CONT. (PAC) AIR SAMPLING SYST. FLOW. TR.

Summary:

The PAC provides calibration readout during normal plant operation to verify the air sampling system line is open and available.

Safety Evaluation:

This PC/M is not safety related, it adds flow indication to the existing PAC air sampling line. Therefore, the probability of an accident previously evaluated in the FSAR is unchanged. Since it is not connected to any safety related system the consequences of an accident previously evaluated in the FSAR are no greater.

PLANT CHANGE/MODIFICATION 86-067

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 3/9/87

SUMMARY DATE: 4/2/87

REVISION: 0

TURBINE AUXILIARIES - BLOCKING OF AUTO-LOADING ON DIESEL GENERATORS

Summary:

This PC/M modified the control circuits of the Turning Gear Oil Pump, Bearing Oil Lift Pump and Turning Gear Motor to prevent auto loading on the EDG on a loss of offsite power; and allow automatic starting only when offsite power is available. This was done to achieve better load management during operation of the EDG'S.

Safety Evaluation:

This PC/M is nuclear safety related since it involves 4A & 4B sequencers. An unreviewed safety question is not involved since the turning gear system is non-class 1E and the ability to manually start the equipment is maintained. Additionally the operability of the EDG is increased by reducing the total load automatically sequenced on following a Loop. This modification does not reduce the integrity, operation, or function of any safety related system addressed in the Tech. Specs.

PLANT CHANGE/MODIFICATION 86-005

PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 06-09-86

SUMMARY DATE: 04-13-87

REVISION: 0

MSIV NITROGEN SUPPLY ADDITION FOR UNIT 3 (INTERIM)

Summary:

The PC/M provides an interim system to supply nitrogen to the 3 Main Steam Isolation Valves. This system is manually actuated and serves as a backup to the existing Instrument Air Supply, to ensure that the MSIV's could be maintained closed in the event of a small steam break downstream of the MSIV's and the Instrument Air System is inoperable.
This system is designed in accordance with respect to FSAR parameters and NRC concerns on valve closure.
The interim system does not, however, address the specific requirements to achieve a second MSIV closure, single failure and other operability concerns; these will be addressed in the final (and permanent) Nitrogen Backup System, Desgin.

Safety Evaluation:

This temporary change is the first step to upgrade the MSIV's to meet the FSAR closure requirements of five second closure with no steam flow and loss of Instrument Air. The operating limits on the Instrument Air System have been specified, which permits power operation of the plant. The change does not affect the operability of the Instrument Air System with respect to closing the MSIV's.
It is noted that the Nitrogen Supply System operability must be verified when the system is placed in service (i.e. when air press <66 psig. and MSIV will not close) in order to ensure that the MSIV'S can be closed in the event of a postualted accident.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 85-130

PC/M CLASSIFICATION: NSR

UNIT: 3

IMPLEMENTED: 3/9/87

SUMMARY DATE: 5/11/87

REVISION: 0

AFW DISCHARGE FLOW CONTROL VALVE UPGRADE

Summary:

This PC/M modifies the AFW flow control valve trim to provide better controlability at the present design and operating conditions. The existing valves have experienced "oscillations" due to the valve operating outside the desirable controlability range of the trim.

Safety Evaluation:

This modification is consistent with all applicable design requirements for the AFW system and will resolve the flow oscillation problem. Appropriate testing criteria is provided for verifying the acceptability of the modifications.

Therefore, this modification does not involve an unreviewed safety question and is considered acceptable.

PLANT CHANGE/MODIFICATION 85-149 PC/M CLASSIFICATION: QUALITY RELATED-NNSR

UNIT: 3

TURNED OVER DATE: 3/9/87

SUMMARY DATE: 5/11/87

REVISION: 0

SPENT FUEL POOL AIR INLET DAMPER REPLACEMENT

Summary:

The engineering package provided for the replacement of the two existing air inlet dampers, integral roughing filters and damper actuators of the Spent Fuel Pool Area Ventilation System. The replacement dampers are Pathway Parallel Multi-Blade dampers provided with integral filters and motorized damper actuators.

The original dampers and integral roughing filters exhibited corrosion around the supporting frames. Examination revealed that the supporting frames and dampers were only partially operable and the filter housing was beyond repair.

Safety Evaluation:

The dampers, damper mountings, damper actuators and associated raceways have been designed as Quality Related in accordance with FSAR Appendix 5A requirements, to prevent interaction with safety related equipment or function.

The installation of the replacement dampers will be performed in a controlled manner under existing, approved plant procedures. The SFP Air Inlet Dampers do not interface with safety related equipment or perform safe shutdown functions, and do not adversely affect system operation. Therefore, the consequence of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

Based on the above evaluation and information supplied by design analysis, it can be concluded that the modification specified in this PC/M does not require a change to any Technical Specification nor does it constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 82-036

PC/M CLASSIFICATION: NNSR-QA/QC

UNIT: 4

TURNED OVER DATE: 5/15/86

SUMMARY DATE: 5/11/87

REVISION: 0

SPENT FUEL PIT LEVEL INDICATOR AND ALARM

Summary:

PC/M 82-36 addressed the installation of the new "SFP Level Alarm System". The System consists of a level transmitter and level indicator supported by structural steel members attached to the spent fuel pool floor. Other equipment consists of a receiver support, control panel, and inputs to the control room.

The system provides continuous SFP level indication locally with inputs to the high and low level alarms.

Safety Evaluation:

The SFP level alarm system is not safety related per the Power Plant Engineering I&C Section, and failure of any of the systems structural supports will not adversely affect any safety related systems or components.

The SFP level indicator does not perform a nuclear safety related function.

The probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to the safety of the plant, previously evaluated in the FSAR, has not been increased. There is no possibility of accident or malfunction different than those previously evaluated. Therefore, it can be concluded that this PC/M does not pose any unreviewed safety questions.

PLANT CHANGE/MODIFICATION 84-167

PC/M CLASSIFICATION: NNSR

UNIT: 3 & 4

TURNED OVER DATE: 3/26/87

SUMMARY DATE: 5/11/87

REVISION: 0

3/4 DECONTAMINATION SHOWER FACILITY

Summary:
A shower facility was added next to the access and dress facility near the auxiliary building. The facility provides a means of showering personnel for the purpose of decontamination. The area has filtered ventilation and sump which prevents the contaminant from escaping to the environment.

Safety Evaluation:
This shower facility does not provide a safety function or provide protection for safety related systems or equipment. The building is located away from all safety related structure systems and components. This modification does not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR and does not reduce the margin of safety as defined in the basis for any technical specification

dc/F2

PLANT CHANGE/MODIFICATION 86-158

PC/M CLASSIFICATION: NNSR

UNIT: 3/4

TURNED OVER DATE: 11/18/86

SUMMARY DATE: 5/11/87

REVISION: 0

UNIT 3/4 THROWBOLT REPLACEMENT ON INTAKE STRUCTURE BAYS HATCHWAY

Summary:

U-Bolts and padlocks were installed to replace the throwbolts on the intake hatchcovers east of the travelling screens. This modification was necessary to ensure security from infiltrators.

Safety Evaluation:

This modification replaced the throwbolts on the intake hatchways with U-bolts and padlocks. There was no impact on the intake concrete or hatch structure. Therefore this modification did not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, did not create the possibility of an accident of malfunction of a different type than any evaluated previously in the FSAR and did not reduce the margin of safety as defined in the bisis for any technical specification.

dc/F2

PLANT CHANGE/MODIFICATION 86-166 **PC/M CLASSIFICATION:** QUALITY RELATED, NNSR

UNIT: 3

TURNED OVER DATE: 3/17/87

SUMMARY DATE: 5/11/87

REVISION: 0

FUEL TRANSFER SYSTEM CABLE DRIVE MODIFICATION - DISASSEMBLY OF PRESENT SYSTEM

Summary:

This Engineering Package directed the disassembly of the Unit 3 air motor drive fuel transfer system. The new cable drive system was installed under PC/M 85-53, Fuel Transfer System, Cable Drive Modification - New System Installation, which replaced the old system.

Implementation of this modification, together with PC/M 85-53 improved the reliability of the fuel transfer system and, therefore, reduced maintenance activities, which occasionally affect the critical path for refueling outages.

Safety Evaluation:

The disassembly of the Unit 3 fuel transfer air operated motors and chain drive systems does not have any direct effects on nuclear safety since the fuel transfer system is not safety related. It could, however, indirectly affect the function of the new transfer system components, and of the transfer and refueling canal liner plates, if these components are damaged during the disassembly effort. As discussed in the Design Analysis, QC inspections of these components will be required after the completion of the disassembly effort to insure that no damage has occurred.

Based on the above, the probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to the safety of the plant has not been increased. There is no possibility of an accident or malfunction different than those previously evaluated in the FSAR. Also, the margin of safety as defined in the Plant Technical Specifications has not been reduced. Therefore, it can be concluded that these modifications to the fuel transfer system do not pose an unreviewed safety question pursuant to 10 CFR 50.59.

PLANT CHANGE/MODIFICATION 86-207

PC/M CLASSIFICATION: QUALITY RELATED-NNSR

UNIT: 3

TURNED OVER DATE: 3/19/87

SUMMARY DATE: 5/11/87

REVISION: 0

IST GAUGE INSTALLATION FOR THE SPENT FUEL PIT COOLING PUMPS

Summary:

The engineering package installed instrumentation for Inservice Testing of the Unit 3 Spent Fuel Pit Cooling Pumps. Specifically it installed a flow element and flow gauge for flow measurement and pump suction and discharge gauges for pressure measurement. This was repaired because these pumps nave been added to the Turkey Point Inservice Testing Program.

Safety Evaluation:

The flow element added by this engineering package is classified as Quality Related for seismic considerations. The flow and pressure gauges are Not Nuclear Safety Related. None of these instruments perform a safety function.

The work associated with this modification is Quality Related. This modification does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-159 PC/M CLASSIFICATION: NNSR

UNIT: 3 & 4

TURNED OVER DATE: 11-8-86

SUMMARY DATE: 5-11-87

REVISION: 0

3/4 INTAKE STRUCTURE WOOD GRATING LATCHES

Summary:
This modification installed locking devices on the east ends of the wood grating covers just east of the travelling sreens at the 16' elevation. The covers were required to be lockable for security reasons.

Safety Evaluation:
This change installed a locking device to the existing wood grating covers. The installation did not affect the structural integrity of either the concrete slab of wood grating. Therefore, this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, and did not create the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR and did not reduce the margin of safety as defined in the basis for any technical specifications.

dc/F2

PLANT CHANGE/MODIFICATION 86-124

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 11/26/86

SUMMARY DATE: 7/1/87

REVISION: 0

REPLACE OF HIGH RANGE GAMMA RADIATION READOUT MODULE

Summary:

The PCM provided for design modification to the Containment high range gamma radiation readout modules for Ra T-4 6311A and B for Unit 4, located in the control room on vertical panels 4081, 82, these readout modules were not capable of full scale deflection (10^8 R/HR) as required for the Channel check and channel fuction test outlines. The modifications to the readout modules provided the capability of operating within the range of 10^0 to 10^8 R/HR. The modifications were internal to the equipment and do not effect original qualifications to the inherent function, or design basis of the system.

Safety Evaluation:

The entire modification was nuclear safety related. The changes were very minor and did not constitute an unreviewed safety question and is considered acceptable.

PLANT CHANGE/MODIFICATION 83-140

PC/M CLASSIFICATION: NNSR

UNIT: 3 & 4

TURNED OVER DATE: 12/01/86

SUMMARY DATE: 7/1/87

REVISION: 0

FIRE DETECTION FOR APPENDIX R MODIFICATIONS

Summary:

This PC/M installed a new low voltage fire detection system in vital areas of the plant to meet the requirement of appendix R.

Safety Evaluation:

This PC/M is not safety related and has a minimum interface with S.R. equipment also was implemented in a controlled manner. It will improve the reliability of the system, therefore, the probability of the occurrence of an accident previously evaluated in the FSAR will not increase. The consequences of an accident previously evaluated in the FSAR will not increase.

PLANT CHANGE/MODIFICATION 84-124

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 3/9/87

SUMMARY DATE: 4/8/87

REVISION: 0

UNIT 4 AUXILIARY FEEDWATER FLOW TRANSMITTER REPLACEMENT

Summary:

This PC/M replaced Unit 4's Barton ITT-752 AFW flow transmitters with Rosemount 1153 flow transmitters. Transmitters replaced were FT-4-1401 A & B, FT-4-1457A & B and FT-4-1458A & B.

Safety Evaluation:

This PC/M is safety related; the new transmitters are fully qualified. The probability of occurrence, or the consequences of a design basis accident, or malfunction of equipment important to safety, as previously evaluated in the FSAR, will not be increased.

PLANT CHANGE/MODIFICATION 82-311

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 3/13/87

SUMMARY DATE: 7/1/87

REVISION: 0

AUXILIARY FEEDWATER TURBINE STEAM SUPPLY STOP/CHECK VALVE.

Summary:

This modification consisted in replacing existing Walworth valves for the auxiliary feedwater pump valves with ones from Pacific Valve Company. The original valve manufacturer could not provide replacement valves that met current required specifications.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since valves with original specifications were installed, with no change in probability of malfunction/accident previously analyzed in the FSAR. The margin of safety was not decreased as previously analyzed in the FSAR.

PLANT CHANGE/MODIFICATION 85-143

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 6/13/87

SUMMARY DATE: 7/1/87

REVISION: 0

BREAKER/FUSE COORDINATION MODIFICATIONS

Summary:

This PC/M changed the power supplies to the 120 V AC vital subpanels from a vital panel breaker to the vital panel main breaker. It replaced the 480V SWGR BKRS. 3(4)0112, 3(4)0206, 3(4)0306, 3(4)0407, 30406 & 40311 trip settings from 1000 amps. to 800 amps.

Safety Evaluation:

This PC/M is safety related. It does not involve an unreviewed safety question since the reliability of the 120V AC vital panels & subpanels and 480 V load center breakers does not change or affect the design basis function of the existing plant systems.

PLANT CHANGE/MODIFICATION 85-60

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 6-11-87

SUMMARY DATE: 7-1-87

REVISION: 0

MAIN TRANSFORMERS FAN COOLER UPGRADE

Summary:

This PC/M added 10 fans for additional cooling capacity to the main transformer.

Safety Evaluation:

This PC/M is not safety related. This modification involves the one for one replacement of original fan cooler units with improved fan cooler units, and does not affect any S.R. equipment. Therefore, no unreviewed safety question is involved.

PLANT CHANGE/MODIFICATION 84-171

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 2/2/87

SUMMARY DATE: 7/1/87

REVISION: 0

MODIFICATIONS TO ALLEVIATE SHORTAGE OF COMPARTMENTS ON MCC 3B

Summary:

This PCM relocated the feeder from MCC 3B (3B06) to load center 3B (3B02) for normal containment cooler fan 3B.

Safety Evaluation:

This PC/M is safety related, this change does not alter the reliability of the normal containment cooling fan 3B, the probability of occurrence of an accident previously evaluated in the FSAR is not increased. Also changes do not constitute an unresolved safety question.

PLANT CHANGE/MODIFICATION 86-236

PC/M CLASSIFICATION: NNSR

UNIT: 3 & 4

TURNED OVER DATE: 4/4/87

SUMMARY DATE: 7/1/87

REVISION: 0

REPLACEMENT OF TREATED WATER PUMP SEALS

Summary:

The treated water pump (P-18A AND P-18B) packing was replaced with mechanical split seals to eliminate the excessive leakage and required maintenance.

Safety Evaluation:

This change was made to non-safety related components and did not alter or impact any safety related components or systems. No unreviewed safety question existed as a result of this modification.

PLANT CHANGE/MODIFICATION 85-197

PC/M CLASSIFICATION: NNSR

UNIT: 3

TURNED OVER DATE: 9/26/86

SUMMARY DATE: 7/1/87

REVISION: 0

RELOCATION OF INSTRUMENT AIR SUPPLY VALVES 40-4-098 AND 40-3-641.

Summary:

This modification relocated valves 40-4-098 and 40-3-641, which supply instrument air to the Unit 3 MFW and AFW flow control valves. This relocation was needed to make these valves more accessible.

Safety Evaluation:

This modification was evaluated and it was concluded that no unreviewed safety question existed as a result of the modification, primarily because the only changes were the physical location of the valves, and the new locations were more accessible to the operators.

PLANT CHANGE/MODIFICATION 85-170

PC/M CLASSIFICATION: SR
UNIT: 3
TURNED OVER DATE: 1/15/87
SUMMARY DATE: 7/1/87
REVISION: 0

INSTALLATION OF AFW VALVE ACCESS PLATFORMS

Summary:

This PCM provided two new access platforms with their associated component lighting for the auxiliary feedwater flow control valves (CV-3-2831, 2832, and 2833) and manual isolation valves (3-141, 241, 341; AFPD-3-006, 007 and 008) which were not easily accessible.

Safety Evaluation:

The platforms do not perform safety related functions and are not associated with safety related systems. However, they are in the vicinity of safety related equipment. Therefore, they have been designed for seismic II/I considerations, hurricane, and tornado wind loads. In addition, they are protected from externally generated missiles due to their location under Unit 3 feedwater platform. the design neither creates a new condition nor altered an existing condition in the plant which has not been analyzed in the FSAR. Therefore, the design did not create an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-067

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 3/9/87

SUMMARY DATE: 4/2/87

REVISION: 0

TURBINE AUXILIARIES - BLOCKING OF AUTO-LOADING ON DIESEL GENERATORS

Summary:

This PC/M modified the control circuits of the Turning Gear Oil Pump, Bearing Oil Lift Pump and Turning Gear Motor to prevent auto loading on the EDG on a loss of offsite power; and allow automatic starting only when offsite power is available. This was done to achieve better load management during operation of the EDG'S.

Safety Evaluation:

This PC/M is nuclear safety related since it involves 4A & 4B sequencers. An unreviewed safety question is not involved since the turning gear system is non-class 1E and the ability to manually start the equipment is maintained. Additionally the operability of the EDG is increased by reducing the total load automatically sequenced on following a Loop. This modification does not reduce the integrity, operation, or function of any safety related system addressed in the Tech. Specs.

PLANT CHANGE/MODIFICATION 87-086

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 3/28/87

SUMMARY DATE: 6/18/87

REVISION: 0

PERSONNEL AIRLOCK EQUALIZATION VALVE REPLACEMENT

Summary:

The Unit 4 Personnel Hatch air lock equalization valve was replaced with a valve determined to be identical with the exception of the valve stems. This replacement valve stem was modified to match the original valve stem prior to installation. After installation, the new valve was tested satisfactorily.

Safety Evaluation:

This mosification is nuclear safety related. Thw replacement valve and linkage meet the original design requirements and function in the same manner as the original. This modification did not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility for an accident or malfucntion of a different type than any evaluated provIously in the FSAR and does not reduce the margin of safety as defined in the basis for any technical specification.

PLANT CHANGE/MODIFICATION 87-095	PC/M CLASSIFICATION: SR
	UNIT: 4
	TURNED OVER DATE: 5/14/87
	SUMMARY DATE: 6-18-87
	REVISION: 0

4 A ICW PUMP ANCHOR BOLTS REPLACEMENT

Summary:

The Southeast and Southwest anchor bolts for the 4A ICW pump base were replaced. The bolts were replaced with through bolts of a larger diameter and a plate on the lower side, which provide greater anchorage than the originally designed anchor bolts. In the process of changing the anchor bolts, some of instrument tubing used to measure differential pressure of the travelling screens was routed to avoid interference with the through bolt.

Safety Evaluation:

This anchor bolt replacement is Nuclear Safety Related since the ICW pumps are safety related. The anchor bolts installed by this CPWO provide a greater anchoring capacity than the originally designed and installed anchor bolts. The new anchor bolts do not affect the operation of the ICW pump. The instrument tubing to the travelling screen differential pressure indicator was not affected by the rerouting of the tubing. Therefore this modification did not increase the possibility of occurrence of the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility for an accident or malfucntion of a different type than any evaluated provіously in the FSAR and does not reduce the margin of safety as defined in the basis for any technical specification.

PLANT CHANGE/MODIFICATION 84-121 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 3/30/87

SUMMARY DATE: 7/7/87

REVISION: 3

UPENDER LEVELING DEVICE MODIFICATION - UNIT 3

Summary:

Structural Modifications were made to the Unit 3 Refueling Pool Upender Leveling Device Bracket Attachment including Bracket shim, Plate removal, Bracket modification, and the addition of stiffener plates to the existing Embed Plate.

Safety Evaluation:

This Modification does not involve safety related Snubbers, does not change SFP operation, and does not affect Block Walls. It will also not cause or be adversely affected by flooding. This Modification does not alter the design bases or function of the upender leveling device.
This change does not constitute an unreviewed safety question. It does not change or increase the probability of an accident previously evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 86-029 PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 3/28/87

SUMMARY DATE: 7/7/87

REVISION: 0

AFW LOCAL INDICATION UNDER THE MAIN FEEDWATER PLATFORM

Summary:

This PCM provided for the installation of flow indicators for AFW Trains 1 and 2, and Steam Generator wide range level indicators under the Main Feedwater Platform for use during manual operation of the AFW Train 2 Flow Control Valves for Unit 4, in case of a Control Room evacuation. In addition, existing flow indicators located on the Main Feedwater Platform were replaced to maintain system compatibility and reliability.

Safety Evaluation:

The previous design requires that plant operations depend on radio communication when manually operating the Train 2 Flow Control Valves. Operation of these valves is required during Control Room evacuation. Installation of new flow and level indicators will help preclude possible errors during a Control Room evacuation and will increase overall reliability of manual system operation and operator action. The modification under this PCM does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-021

PC/M CLASSIFICATION: SR

UNIT: 3 & 4

TURNED OVER DATE: 4/13/87

SUMMARY DATE: 7/7/87

REVISION: 0

"C" AUXILIARY FEED PUMP REPLACEMENT IMPELLER

Summary:
This CPWO replaces the currently installed rotating element in the Auxiliary Feedwater Pump C with the spare rotating element. The impeller vanes of the spare rotating element have been underfiled in the vendor shop in order to increase the head of the Auxiliary Feedwater pump. The replacement rotating element is identical to the previously installed element except for the underfiled impeller vanes.

Safety Evaluation:

The replacement of the pump impeller assembly (rotating element) does not involve an unresolved safety question because the probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased, the possibility for an accident or malfunction of a different type than evaluated previously in the FSAR is not created, and the margin of safety as defined in the basis for a Technical Specification is not reduced.

PLANT CHANGE/MODIFICATION 86-212

PC/M CLASSIFICATION: SR

UNIT: 3 & 4

TURNED OVER DATE: 2-12-87

SUMMARY DATE: 7-7-87

REVISION: 0

ENVIRONMENTAL QUALIFICATION LIST REVISION

Summary:

The purpose of this PC/M was to correct EQ list deficiencies and update EQ document packages. No hardware change was involved.

Safety Evaluation:

The PC/M was designated SR because the EQ list and associated Doc. Packs are SR. This PC/M did not make a physical change to the plant. No technical specification change was required. The PC/M does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-103	PC/M CLASSIFICATION: SR
	UNIT: 3 & 4
	TURNED OVER DATE: 2/12/87
	SUMMARY DATE: 7/7/87
	REVISION: 0

ENVIRONMENTAL QUALIFICATION LIST REVISION

Summary:

The PCM updated EQ List based upon the EQ update EQ Doc Packs and the associated review of PC/M'S for the purpose of identifying equipment and components within the scope of 10 CFR 50.49. The PCM only revised drawing 5610-E-1435 rev 1 (EQ LIST) and involved no plant Mod's.

Safety Evaluation:

The PCM is S.R. but did not constitute an unreviewed safety question, did not require any changes to TECH. SPEC'S and therefore is considered acceptable.

PLANT CHANGE/MODIFICATION 85-65

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 1/15/87

SUMMARY DATE: 7/7/87

REVISION: 0

G.E. SAM RELAY MOD., P.C. CARD REPLACEMENT

Summary:

This PC/M replaced a printed circuit card on existing SAM Relays to eliminate the possibility of incorrect timing due to initiating contact bounce.

Safety Evaluation:

This PC/M is safety related and does not involve an unreviewed safety question because it involves the replacing of a P.C. with a slightly modified card. No external wiring changes were made and the relay's function remain unchanged.

PLANT CHANGE/MODIFICATION 83-153

PC/M CLASSIFICATION: SR

UNIT: 3 & 4

TURNED OVER DATE: 12/18/86

SUMMARY DATE: 7/7/87

REVISION: 0

CABLE REROUTING - APPENDIX R MODS. UNITS 3 & 4.

Summary:

This PC/M re-routed cables in conduits to ensure hot and/or cold shutdown capability and for which protection in place is not practical to meet section III G.2 of appendix R.

Safety Evaluation:

This PC/M is safety related, the probability of an accident previously evaluated in the FSAR will not be increased as there are no additional changes other than the pulling and termination of new cables in new raceways which are seismically supported.

PLANT CHANGE/MODIFICATION 87-023

PC/M CLASSIFICATION: QR

UNIT: 4

TURNED OVER DATE: 6/19/87

SUMMARY DATE: 7/7/87

REVISION: 0

MAIN STEAM HYDRAULIC SNUBBER REPLACEMENT

Summary:

PC/M # 87-023 provided for the replacement of certain hydraulic snubbers in the Main Steam System for Turkey Point Unit 4. The affected supports for this snubber replacement effort were as follows:

* - MSHX-1
* - MSHX-2
* - MSHX-4A
* - MSHX-4B
* - MSHX-7
* - MSHX-8
* - MSHX-10

The replacement of these snubbers was necessitated by a history of problems and poor performance of this particular type of hydraulic snubber. PC/M # 87-023 replaced these hydraulic snubbers with Anchor/Darling mechanical snubbers manufactured in accordance with ASME Section III, Subsection NF which were determined to be equal to or better than the original hydraulic snubbers. The replacement mechanical snubbers provide maintenance free operation, a capacity to test in place and a suitability for operation in the vibration environment of the Main Steam piping.

Safety Evaluation:

The Hydraulic snubbers that were replaced by PC/M # 87-023 are classified as Quality Related and as such are not addressed in the Turkey Point Technical Specification, Sections 3.13 and 4.14. Furthermore, the current FSAR and Technical Specification specify snubber examination and test requirements only on safety related snubbers.

However, the replacement of the subject hydraulic snubbers in the Main Steam System with mechanical snubbers that are equal to or better than the existing snubbers does not affect nuclear safety. This modification is considered a one-for-one replacement and therefore does not impact on existing safety related functions, create malfunctions of a different type than previously evaluated in the safety analysis report or reduce the margin of safety for any technical specification.

PC/M 87-023 is provided with a written safety evaluation in accordance with 10 CFR 50.59. The safety evaluation concludes that the change performed under the PCM does not involve an unreviewed safety question, consequently prior commission approval for the implementation of the modification was not required.

PLANT CHANGE/MODIFICATION 86-96

PC/M CLASSIFICATION: SR
UNIT: 4
TURNED OVER DATE: 9-8-86
SUMMARY DATE: 7-7-87
REVISION: 0

NEW POWER FEED TO NON-VITAL SECTION OF MCC 4A

Summary:

PCM eliminated the consequences of a failure of MCC 4A non-vital to vital tie breaker to trip during EDG load sequencing.

Safety Evaluation:

Per the existing plant design, the non-vital section of MCC 4A is fed from the vital section of MCC 4A through a tie breaker, and is automatically shed after a time delay on a loss of offsite power by tripping of the tie breaker.. By implementation of this PCM, the power feed for MCC 4A non-vital bus is relocated from the vital bus of MCC 4A to the source Load Center 4A, and the existing tie breaker will be used as the incoming breaker at the MCC non-vital bus NV4A. NV4A will be shed on loss of load center bus voltage, with no intentional time delay, by independent trip circuits to the new load center Feeder Breaker 40103 and the MCC incoming Feeder Breaker 40535 is required by 10CFR50.59, an evaluation has been performed to determine if this modification constitutes an unreviewed safety question and requires prior NRC approval. The results are as follows.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/-accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-060

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 2-20-87

SUMMARY DATE: 7-7-87

REVISION: 0

COMPUTER ROOM TEMPERATURE INDICATION

Summary:

This PCM adds two redundant temperature indication loops to enable the control room operators to monitor the Computer room temperature. The operators need this information to determine when to restart the chillers and HVAC equipment following a loss of offsite power. Thus, the load management program for the diesels will be enhanced by not adding unnecessary KW.

Safety Evaluation:

The chillers and air handling equipment associated with the Computer Room do not automatically reload onto the vital AC bus after SI and/or LOOP. Since the Computer Room houses safety related equipment (ICCS cabinets) which have finite temperature requirements, the operator must manually restart the HVAC before reaching the temp. limit. This PC/M, therefore, does not constitute an unreviewed safety question; it provides the operator with vital information to prevent violation of operating limits.

PLANT CHANGE/MODIFICATION 85-071 PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 4-15-87

SUMMARY DATE: 7-7-87

REVISION: 2

SPENT FUEL PIT BUILDING WALL JOINT REPAIR

Summary:

This modification provided for the repair of expar-ion joints in the South wall of the SFP Bldg., and in the South Door Column in the East wall. The existing material was replaced by an epoxidized polyurethane sealer. The new joint material is a "Dymeric" Sealant, backed by a polyethelene Rod.

Safety Evaluation:

The new joint material does not perform a safety related function the installation, on failure of this new material will not affect any safety related system. This does not pose an unreviewed safety question pursuant to 10CFR 50.59.

PLANT CHANGE/MODIFICATION 85-35

PC/M CLASSIFICATION SR

UNIT: 3

TURNED OVER DATE: 6/19/87

SUMMARY DATE: 7/7/87

REVISION: 0

REPLACEMENT OF PYCO RTD'S

Summary:

This PC/M replaces PYCO RTD's used in the RCS and charcoal filters with CONAX RTD'S. The PYCO RTD'S had reached the end of their qualified life thus necessitating replacement.

Safety Evaluation:

This PC/M did not change the function of the SR RCS RTD'S. The replacement CONAX RTD'S are fully qualified. No other SR equipment was affected by this PC/M. This PC/M did not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-97

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 6/27/87

SUMMARY DATE: 7/13/87

REVISION: 0

INSTALLATION OF UV TRIP CIRCUITRY FOR TURBINE & POLAR CRANE BREAKERS

Summary:

This PC/M installed trip coils & associated circuitry in breakers for Turbine Crane & Polar Crane. These will trip the breakers when an undervoltage condition is sensed on the affected 480V bus. This is to ensure that the loads will be disconnected from the Emergency Diesel Generators following a loss of offsite Power.

Safety Evaluation:

This PC/M is safety related. It does not involve an unreviewed safety question since it does not add or change any equipment required to operate during an accident & it does not involve any Technical Specification. Additionally, it improves the margin of safety of the EDG's by minimizing loads that could be auto connected .

PLANT CHANGE/MODIFICATION CPWO 87-52 PC/M CLASSIFICATION: NNSR

UNIT: 3

TURNED OVER DATE: 6/9/87

SUMMARY DATE: 7/20/87

REVISION: 0

UNIT 3 GENERATOR NEUTRAL GROUNDING TRANSFORMER REPLACEMENT

Summary:

This CPWO replaced the existing gerneator neutral grounding transformer which contained PCB's with one that did not contain PCB's. The replacements are manufactured to original physical & operational design requirements

Safety Evaluation:

This CPWO is non-safety related because the transformer is part of the main generator & is dated phase bus system which is non-safety related, and non-essential for safe shutdown. This change does not involve an unreviewed, safety question and no TECH SPEC changes are requires.

PLANT CHANGE/MODIFICATION CPWO 87-53 PC/M CLASSIFICATION: NNSR

UNIT: 4

TURNED OVER DATE: 6/9/87

SUMMARY DATE: 7/20/87

REVISION: 0

UNIT 4 GENERATOR NEUTRAL GROUNDING TRANSFORMER REPLACEMENT

Summary:

This CPWO replaced the existing gerneator neutral grounding transformer which contained PCB's with one that did not contain PCB's. The replacements are manufactured to original physical & operational design requirements

Safety Evaluation:

This CPWO is non-safety related because the transformer is part of the main generator & is dated phase bus system which is non-safety related, and non-essential for safe shutdown. This change does not involve an unreviewed, safety question and no TECH SPEC changes are requires.

PLANT CHANGE/MODIFICATION 87-98

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 6/23/87

SUMMARY DATE: 7/20/87

REVISION: 0

INSTALLATION OF UNDERVOLTAGE TRIP DEVICE FOR POLAR CRANE BREAKER

Summary:

This PC/M installed trip coils & associated circuitry in breaker for Polar Crane. This will trip the breaker when an undervoltage condition is sensed on the affected 480B bus. This is to ensure that the Polar Crane will be disconnected from the Emergency Diesel Generator following a loss of offsite power.

Safety Evaluation:

This PC/M is safety related. It does not involve an unreviewed safety question since it does not add or change any equipment required to operate during an accident & it does not involve any Technical Specifications. Additionally, it improves the margin of safety of the EDG by minimizing loads that can be auto connected.

PLANT CHANGE/MODIFICATION 80-117

PC/M CLASSIFICATION: NNS

UNIT: 3 & 4

TURNED OVER DATE: 11-15-85

SUMMARY DATE: 7/29/87

REVISION: 3

UPGRADE AUXILIARY FEEDWATER SUCTION, DISCHARGE AND STEAM SUPPLY PIPING

Summary:

This modification consisted of adding redundant steam supplies to the AFW trubines. The modification also replaced the auxiliary feedwater control valves and removed the following lines from the condensate storage tank discharge line: condensate makeup/reject line, condensate recovery system discharge line and condensate transfer pump line.

Safety Evaluation:

This change is safety related but does not involve an unreviewed safety question as this modification does not affect, create or increase the probablity of occurrence of any accident/malfunction already addressed, or new, in the FSAR.

PLANT CHANGE/MODIFICATION 84-111

PC/M CLASSIFICATION: NNS = QA/QC

UNIT: 3 & 4

TURNED OVER DATE: 10/31/86

SUMMARY DATE: 07/29/87

REVISION: 0

A/C UNIT FOR PASS CONTROL PANEL

Summary:

This PC/M provided an air conditioner for P.A.S.S. control panel CZ14 in the auxiliary building hallway outside of the P.A.S.S. room. The air conditioner was added in order to prolong the life and increase the relaibility of electrical components in the cabinet.

Safety Evaluation:

The addition of the air conditioner to Cabinet C-214 does not increase the probability of occurrence of any accident previously evaluated in the FSAR, nor will it affect the consequences of any accident previously evaluated in the FSAR. The P.A.S.S. Control Cabinet has no safety related function and the failure of the air conditioner has no potential for interaction with safety related equipment. The modification will not affect the consequences of malfunction of equipmernt important to safety previously evaluated in the FSAR, and the possibility of an accident of a different type than any analyzed in the FSAR is not created by this PC/M. The margin of safety as defined in the basis for any Technical Specification is not affected by this PC/M.

PLANT CHANGE/MODIFICATION CPWO 86-15 PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 10-13-86

SUMMARY DATE: 07-29-87

REVISION: 0

REPLACEMENT OF TELEDYNE-FARRIS CCW RELIEF VALVES

Summary:

CPWO 86-15 allowed obsolete, Teledyne-Farris model 1870, relief valves that failed periodic testing to be replaced by Teledyne-Farris model 1850 valves. The model 1850 is identical to the 1870 except for minor internal changes. All specifications of the 1870 valve are met by the replacement.

Safety Evaluation:

This nuclear safety related CPWO provided for component cooling water system relief valve replacement. The replacement relief valves are functionally identical to the original valves. Therefore, implementation of this CPWO did not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-068

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 10/31/86

SUMMARY DATE: 07/29/87

REVISION: 0

REMOVAL OF CCW PIPING TO THE PRIMARY SHIELD COOLERS

Summary:

PCM 86-068 removed the Component Cooling Water Piping valves, instrumentation and associated hardware for the Primary Shield Coolers. The coolers were originally installed to maintain temperatures within the primary shield wall below 150°F, however, the design of the shield wall did not take credit for cooling by the primary shield coolers. Since the coolers were not required and were in need of maintenance it was decided to abandon them by performing the work described in this PCM.

Safety Evaluation:

PCM 86-068 is considered nuclear safety related since it removes portions of the component cooling water system associated with the primary shield coolers. This PCM does not adversely effect any other function of the component cooling water system. The primary shield coolers were not required for either normal plant operation or post accident recovery. Therefore, implementation of this PCM did not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-169 PC/M CLASSIFICATION: SAFETY RELATED

UNIT: 4

TURNED OVER DATE: 6/19/87

SUMMARY DATE: 7/29/87

REVISION: 0

MODIFICATION TO COMPONENT COOLING WATER SYSTEM

Summary:

Plant Change/Modification 87-169 covers modifications to the pipe supports on the CCW lines to and from the Reactor Coolant Pump Thermal Barriers. These improvements were made to minimize the possibility of damage to the CCW lines if a thermal barrier were to fail. This modification was not a direct NRC commitment or requirement.

Safety Evaluation:

The section of Component Cooling Water Pipe that is being modified by this PC/M is safety related. An unreviewed safety question is not involved with these modifications. The changes in this PC/M are to pipe supports and do not change the system operation, create malfunctions of a different type than that evaluated in the safety analysis report or reduce the margin of safety.

PLANT CHANGE/MODIFICATION 81-059 PC/M CLASSIFICATION: NNSR

UNIT: COMMON

TURNED OVER DATE: 2-17-87

SUMMARY DATE: 7-30-87

REVISION: 0

WATER TREATMENT PLANT FINAL EFFLUENT CONDUCTIVITY TRIP

Summary:

Under PC/M 81-059, an automatic isolation valve in the common Water Treatment Plant (WTP) discharge line to Units 3 & 4 was installed. The automatic isolation valve was provided with a full flow bypass with appropriate isolation valves to permit maintenance of the automatic isolation valves to permit maintenance of the automatic isolation valve. A handswitch ,with valve open and valve closed position indication lights, was provided in the WTP control panel to permit operators to remotely open and close the automatic isolation valve.

In addition to the above, a conductivity cell was provided upstream of the automatic isolation valve in order to monitor WTP final effluent, conductivity. The design is such that if effluent conductivity levels , as sensed by the newly installed conductivity cell, increase to an established setpoint then the following ocur:

- The automatic isolation valve closes
- The demineralizer feed pumps in the WTP trip
- An annunciation in the WTP control panel annunciates

Safety Evaluation:

PC/M 81-059 involved ohly changes to the Water Treatment Plant (WTP), a non-nuclear safety related system. This system does not function to achieve and maintain safe shutdown condition, or to safely store and cool spent fuel, or to prevent or mitigate accidents, which could result in potential off-site radiological exposurer comparable to those cited in 10 CFR 100.11.

The change performed under PC/M 81-059 does not interface with any safety-related equipment nor is it located in the vicinity of any safety-related equipment. Therefore, failure of it's pipe supports would not adversely affect safety related systems, equipment or structures, and it can be concluded that PC/M 81-059 does not pose any unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-209 PC/M CLASSIFICATION: NNSR

UNIT: 4

TURNED OVER DATE: 11-18-87

SUMMARY DATE: 7/30/87

REVISION: 0

MSR FOUR TUBE PASS MODIFICATION

Summary:

PC/M 83-209 provided for the installation of a Scavenging Steam Vent Condenser (SSVC) drain lines from each Moisture Separator Reheater (MSR) to a HP (No.6) Feedwater Heater or to the Main Condenser. The modifications performed under this PC/M are related to modifications performed under PC/M 85-133 which converted the internals of each MSR from a two-pass tube arrangement to a four-pass tube arrangement for reheating cycle steam. The additional two passes produce excess condensate of the reheating steam which is removed from each MSR via an installed SSCV drain line.

The SSCV lines discharge directly into a HP Feedwater Heater (6A or 6B) via an extraction steam line during normal operation and to the Main Condenser during start-up operation. The SSVC lines also provide a means of venting the MSR'S to purge non-condensable gases.

All changes in steam or condensate flowrates due to the above modifications were found to be acceptable and did not require changes to other existing piping and valves except to previously existing vent lines to the MSR'S. These vent lines were cut and capped under PC/M 83-209 since the SSVC drain lines provide the necessary vent path during startup from the MSR's to the condenser.

Safety Evaluation:

The modification performed by PC/M 83-209 is classified as Non-Nuclear Safety Related. The modification involves only Quality Group D secondary side system components such as the MSR's the HP Feedwater Heater and the Main Condenser. The modification does not involve safety related snubbers, safety related instrument lines or any other components important to safety.

The modification does not affect any limiting condition for operation per Turkey Point Technical Specifications.

The modification does not involve the addition of electrical cable or any changes to existing receways.

The addition of the SSVC drain lines by PC/M 83-209 does not impact high energy line break analyzers already evaluated in the FSAR nor do they affect the flooding analysis as described in the NRC Safety Evaluation Report dated September 4, 1979. In addition, the installation of these lines does not create a new hazard to existing safety related systems or components.

PC/M 83-209 is provided with a written safety evaluation in accordance with 10 CFR 50.59 . The conclusion of the safety evaluation is that the modification does not involve unreviewed safety questions.

PLANT CHANGE/MODIFICATION 85-131 PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 9/17/86

SUMMARY DATE: 7/29/87

REVISION: 0

AUXILIARY DISCHARGE FLOW CONTROL VALVE UPGRADE

Summary:

This PC/M replaced the valve seats, retainers and plugs on the AFW Flow Control Valves CV-4-2816, 2817, 2818, 2831, 2832, and 2833. This change allows better operation at the 125 gpm automatic initiation setpoint. The manual handwheels were also locked to restrict valve stem travel to a maximum of 85 percent of full open. This limits flow to less than 525 gpm in the event a flow control valve fails open.

Safety Evaluation:

This modification improves performance of the system at the revised setpoint of 125 gpm and limits flow if an FCV fails open. This modification does not change the operation of the AFW system and it does not increase the probability or consequences of any accident previously analyzed or different from those analyzed. Therefore this change does not create an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-133 PC/M CLASSIFICATION: NNSR

UNIT: 4

TURNED OVER DATE: 11-18-86

SUMMARY DATE: 07/29/87

REVISION: 0

MSR MODERNIZATION

Summary:

PC/M 85-133 provided a modification package for the Moisture Separator Rehaeters (MSR)s to increase MSR performance. The modification included the replacement of each MSR tube bundle and modifications to moisture separation equipment.

Each MSR tube bundle used for reheating cycle steam was replaced by a Westinghouse tube bundle design. This design increased the total number of tubes inside each MSR, and changed the original two-pass tube arrangement to a four-pass tube arrangement. The excess condensate of the reheating steam that occurs as a result of the two additional passes is removed by a Scavenging Steam Vent Condenser (SSVC) drain line that was installed under PC/M 83-209. Modifications to other moisture separation equipment also enhanced overall MSR performance. These changes included the replacement of the original mesh-type moisture separators with cherron separators. Other modifications were also performed to optimize steam flow distribution and to prevent mositure entrainment within the moisture separator section of the MSR's.
PC/M 85-133 also provided for the installation of two Reheater Drain Tank drain line flow measuring instruments, test connection points and thermo-wells.

Safety Evaluation:

The modification performed by PC/M 85-133 is classified as Non-Nuclear Safety Related. The modification involved only Quality Group D components on the secondary side of the plant. These components include the four MSR's and the two Reheater Drain Tanks. The modification does not involve safety related snubbers, safety related instrument lines or any other components inportant to safety.

The modification does not affect the evaluation of any accident previously performed in the FSAR nor does it create or increase the possibility of any accident not already evaluated in the FSAR

The modification does not affect any limiting condition for operation per Turkey Point Technical Specifications.

The modification does not affect the flooding analysis as described in the NRC Safety Evaluation Report dated September 4, 1979.

PC/M 85-133 is provided with a written safety evaluation in accordance with 10CFR50.59. The conclusion of the safety evaluation is that the modification does not involve unreviewed safety questions.

PLANT CHANGE/MODIFICATION 86-31

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 12/10/86

SUMMARY DATE: 7/29/87

REVISION: 0

AUXILIARY FEEDWATER PUMP CONTROL PANEL WIRING MODIFICATIONS

Summary:

This PCM provided wiring modifications to the Auxiliary Feedwater Pump Control Panels to correct a design deficiency which resulted in the short-circuiting and arcing on remote selector switch contacts in the main control boards. This modification corrected this deficiency by utilizing spare contacts on the existing relays already installed within the control circuit. This also provided modifications to the AFW Pump Trip and Throttle Valve limit switch wiring connections to correct the current OPEN and CLOSED sequence of operations for the position indicating lights. Also the Trip and Throttle Valve positions instead of the Limitorque Motor Operator Positions will now be indicated on the main control board.

Safety Evaluation:

The PCM enhanced operator interface with the AFW pump and trubine control panel, and increased the reliability of the controls. None of these changes increased the probability or consequences of any previously or not previously analyzed accident, and therefore do not result in an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-099

PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 6/26/87

SUMMARY DATE: 7/31/87

REVISION: 0

ICW/CCW BASKET STRAINER REPLACEMENT

Summary:

This CPWO requested the rplacement of the Unit 3 ICW/CCW Basket Strainers due to the extensive corrosion of the existing strainers. The new strainers were fabricated to ASME Sect. VIII Div.1 1983ED-1985 Summer Add.; the existing strainer was built to ANSI B31.1 requirements. The original strainers were built by Zurn Industries, and the replacement strainers were built by Zurn's new owner Hayward Industrial Products, to the same dimensional specifications, and essentially the same strainer body material. The strainers are coated with corroglass 200 epoxy material, while the original strainers were coated with coal-tar epoxy. The replacement strainers can also be equipped with a sacrificial zinc anodes to provide additional corrosion protection.

Safety Evaluation:

All dimensions and materials were provided on a one-to-one basis for the replacement strainers, and the new coating is considered to be an improvement. The maximum specified nozzle loads for the strainer were found to be acceptable. The addition of zinc anodes improves the service life of the strainers.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 87-156

PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 6/16/87

SUMMARY DATE: 8/03/87

REVISION: 0

UNIT 3 ICW BASKET STRAINER ISOLATION VALVE REPLACEMENT - SHAFT/OPERATOR ADAPTER

Summary:

This CPWO was prepared to provide the necessary parts to adapt the new replacement valves with the existing hand operators. The replaced valve shaft is smaller than the original valve shaft, requiring a modification to the Henry Pratt MDT-4 operator.

Safety Evaluation:

The valves are normally open and are closed for maintenance purposes only. These valves are not required to operate to perform any safety function and are safety related for pressure boundary concerns only. The change will not affect the valves operability as attested by the manufacturer, Henry Pratt Co.

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-076 PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 6/3/87

SUMMARY DATE: 7/31/87

REVISION: 0

DIESEL GENERATOR "B" FREQUENCY METER REPLACEMENT

Summary:

This modification involved the replacement of an existing, obsolete frequency meter in the Emergency Diesel Generator B Control Cabinet 4C12, with an upgraded, functionally equivalent frequency meter and removal of the assoicated frequency impedor module also installed in Cabinet 4C12. The new meter with self-contained tranducer has only two wiring terminals versus the old meter, which had three wiring terminals and required the use of a frequency impedor module.
The Control Cabinet 4C12 is listed in the Essential Equipment List, however, this PCM does not impact its safe shutdown capability.

Safety Evaluation:

This modification involved the replacement of an existing frequency meter in the Emergency Diesel Generator B Control Cabinet 4C12, with an upgraded, functionally equivalent frequency meter. The existing frequency impedor also installed in Cabinet 4C12 is to be removed as the new frequency meter does not require the use of a frequency impedor. Although the emergency diesel generator frequency meter does not perform a safety related function, this modification is nuclear safety related to ensure the circuit integrity of existing safety related Control Cabinet 4C12.
This modification was not inside containment, does not involve safety related snubbers, does not involve block walls, does not inpact the spent fuel cooling operations of the plant and does not affect Radioactive Waste Treatment System of the plant.
The modification was reviewed for ALARA requirements based upon the guidance provided in Criteria for ALARA Evaluation per FPL letter JPE-PTPO-84-1239.
The modification accomplished by this PCM did not affect the flooding analysis as described in the NRC Safety Evaluation Report, dated September 4, 1979, because the modification does not introduce a new source of safety related componets which could be affected by flooding.
Equipment or cables associated with this work were not attached to or in proximity of any block walls which have not been previously analyzed to preclude their failure and subsequent damage to adjacent safety related equipment. No new cables required. The existing internal wire was used for connecting the new frequency meter.
The replacement of the frequency meter did not adversely affect the seismic qualification of Panel 4C12 as no structural modifications to this panel are required as the new frequency meter utilizes the same mounting arrangement as the existing frequency meter and the meters are of approximately the same size and weight

Based on the preceding, the following conclusions can be made:

- The replacement of the existing frequency meter in the Emergency Diesel Generator B Control Cabinet 4C12, with an upgraded functionally equivalent frequency meter did not change the design function of the Emergency Diesel Generator System. Therefore, the probability of occurrence of an accident previously evaluated in the FSAR will not be increased.
- The consequences of an accident previously evaluated in the FSAR was not increased because the basic function of the Emergency Diesel.

  Generator System remains the same, and no other safety related systems are adversely affected by this modification.

- This modification did not change the inherent function of any safety related systems. Therefore, there was no possibility that an accident may be created which is a different type than any already evaluated in the FSAR.

- This modification was for the replacement of the obsolete frequency meter with an upgraded version of the frequency meter. Therefore, the probability of occurrence of equipment malfunctions important to safety previously evaluated in the FSAR was not increased.

- All work associated with this modification was accomplished in accordance with approved procedures and final system design will be tested to ensure its proper function and operability. Therefore, consequences of equipment, malfucnction important to safety previously evaluated in the FSAR was not increased.

- This modification did not adversely affect the inherent function or design basis of the systems related to safety; therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the FSAR was not created.

- This modification did not reduce the margin of safety as defined in the bases for any Technical Specification since this modification replaces the obsolete frequency meter with an upgraded model. However, strict adherence to the requirements of the Technical Specifications, Section 3.7.[illegible] shall be observed.

Based on the above, this modification did not constitute an unreviewed safety question and is considered acceptable.

PLANT CHANGE/MODIFICATION 86-80 PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 06/26/87

SUMMARY DATE: 07/31/87

REVISION: 1

S.I. ACCUMULATOR MAKEUP HEADER SEISMIC REPLACEMENT

Summary:

This modification qualifies the piping and support configuration of the Accumulator Makeup Header inside containment to the requirements of the FSAR Appendix 5A seismic design criteria by the installation or modification of supports per Specification 5177-M-56 and Computer Stress Analysis.

This modification is inside containment. It consists of new supports and modifications to some existing supports. The Structural steel required for these modifications will change the heat sink by approximately ten square feet of 1/8- inch thick steel. This is a negligible change to the 48,300 square feet used in the ECCS heat sink analysis (Refer to Table 14.2.4-1 of the FSAR). Therefore, this modification does not alter the ECCS heat sink analysis. This modification does not involve safety related snubbers, or block walls, does not impack the spent fuel cooling operations of the plant, does not involve additions of electrical cable or changes to existing raceways, and does not compromise the Turkey Point Fire Protection program.

This modification resolves the safety concerns addressed in JPE-M-85-029 which evaluated the consequences of a post-LOCA break in the accumulator fill line. The possibility of a break had increased when it was determined that the pipe was not seismically supported.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-090 PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 7/8/86

SUMMARY DATE: 8/03/87

REVISION: 0

ROOT VALVE NO 4-20-698 COMPONENT SUBSTITUTION

Summary:

This CPWO replaced broken 3/4" 1500 psi Rockwell globe valve 4-20-698 (FT-4-497 isolation) in the Feedwater system with an equivalent 3/4" 1500 psi globe valve.

Safety Evaluation:

This CPWO will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR, nor will the possibility of an accident or a, malfunction of a different type than any evaluated previously in the FSAR be created. Additionally, the margin of safety as defined in the basis for any technical specification is not reduced.

PLANT CHANGE/MODIFICATION 85-010 PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 2/18/87

SUMMARY DATE: 8/03/87

REVISION: 0

ADDITION OF FEEDWATER CONTROL VALVES DIRECT POSITION INDICATION

Summary:

In order to provide direct position indication for the Main and By-Pass Feedwater Control Valves, FCV-4-478, 488, 498 and FCV-4-479, 489, 499, respectively, qualified limit switches (NAMCO Type EA-180) and indicating lights were added.
The indicating lights are located in the Control Room Control Console 4 directly above each valve's corresponding controller. A new safety related power supply has been provided for the indicating lights.
Conax electric conductor seal assemblies are also provided to maintain the environmental integrity of the limit switches, due to their being located in a, high energy line break area.

Safety Evaluation:

This modification consists of adding qualified limit switches and indicating lights for the feedwater control valves to meet the requirements of Regulatory Guide 1.97, Rev. 3. In the event of a single failure of Train A, valve position can be determined by : 1) the main feedwater flow indicators, 2) SPDS/SAS, or 3) visual inspection of valve.
The circuitry involved in this modification is for indication only and does not perform any control function. The load added to the safety related power supply by this addition is negligible.
This modification did not involve an unreviewed safety question because:

1.a With respect to the probability of occurrence of an accident previously evaluated in the FSAR:

This addition provided valve position indication only; the existing control functions were unchanged. All conduits have been seismically installed. All modifications utilize qualified safety grade components and were done under plant QC procedures. Addition of new items adds negligible weight and had no effect on the equipment seismic response.
Therefore, the probability of occurrence of an accident previously evaluated in the FSAR is not greater.

1.b With respect to the consequences of an accident previously evaluated in the FSAR:
This modification meets the requirements of Regulatory Guide 1.97, Rev. 3. With the addition of qualified limit switches and indicating lights there is a greater assurance that the valves are in the correct position. The control of the valves was not affected by this

change. Addition of new items adds negligible weight and had no effect on the equipment seismic response.
Therefore, the consequences of an accident previously evaluated in the FSAR was not increased.

1.c With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR:

All Conduits and equipment were seismically installed. The addition of the limit switches on the control valves was wired totally independent of the wiring for the solenoid valves controlling the feedwater valves. There were no inter ties with any other safety related system. Addition of new items adds negligible weight and had no effect on the equipment seismic response.
Therefore, the probability of malfucntion of equipment important to safety is no greater.

1.d With respect to the consequences of malfunction of equipment important to safety previously evaluated in the FSAR.
The addition of qualified indicating lights provides for better operator information. The wiring for this modification had no interaction with any other safety related system. All the equipment was seismically installed.

Therefore, the consequences of malfunction of equipment important to safety previously evaluated in the FSAR were not increased.

2.a With respect to the possibility of an accident of a different type than any analyzed in the FSAR:

The new conduits and equipment were seismically installed. This system provides valve position indication only. Existing valve function and control remained unchanged.

Therefore, the possibility of an accident of a different type than any analyzed in the FSAR was not created.

2.b With respect to the possibility of malfunction of a different type than any analyzed in the FSAR:

The circuitry involved in this modification is independent of the control wiring for each valve. This new system is not connected to any other safety related system. The function of the valves was not changed.

Therefore, the possibility of malfunction of a different type than any analyzed in the FSAR was not created.

3 With respect to the margin of safety as defined in the basis for any Technical Specification:

This addition is not addressed in any Technical Specification.

Therefore, no margin of safety as discussed in the Technical Specifications has been decreased.

These modifications added terminal blocks and indicating lights to the control console. The negligible weight of the new equipment and the new cut-outs did not affect the seismic response of the console.

The weight of the limit switches and brackets being added to each valve are negligible and did not affect the seismic response.

All added conduit was seismically installed per 5177-E-302.

The modifications were not installed on or adjacent to any "block" walls.

PLANT CHANGE/MODIFICATION 84-209

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 08/16/86

SUMMARY DATE: 08/06/87

REVISION: 0

REINSTATEMENT OF POWER MISMATCH WITHOUT AUTOMATIC ROD WITHDRAWAL

Summary:

The Automatic Rod Control System, with power mismatch circuitry was potentially susceptible to undesirable control system operations induced by an adverse environment (i.e., a steam line break inside containment could subject the excore detectors and cables to elevated temperatures which could cause rod withdrawal, if the rods were in the automatic mode prior to a reactor trip). Power mismatch was disconnected from automatic rod control by PC/M 81-13. The possibility of the NIS System initiating a spurious low power signal without causing a reactor trip on negative flux rate could have been eliminated by the removal of automatic rod withdrawal circuit. Because the rod insertion circuit was also eliminated, it is deemed necessary to reinstate automatic rod insertion control circuit. When operating in automatic mode, the automatic rod insertion would occur, if nuclear instrumentation system detects a high power signal (OT-OT).

Safety Evaluation:

This change does not involve an unreviewed safety question, because the modification reinstates the power mismatch circuit associated with automatic rod insertion only. The probability of occurrence of uncontrolled rod cluster control assembly (RCCA) withdrawal is not made more likely, since this modification affects the rod insertion circuitry only and all the rod withdrawal circuitry will be disconnected. The power mismatch circuitry was provided as part of the original NSSS package. This modification only reinstalls the automatic rod insertion circuitry to its original state and removes the circuitry associated with automatic rod withdrawal, since this modification does not add a control system that did not exist. Hence the probability of occurrence of an accident previously evaluated in the FSAR, or consequences of an accident, or probability of malfunction of equipment important to safety, or consequences of the malfunction of equipment important to safety previously evaluated in the FSAR has not changed. Since the modification reinstates a system that was provided in original NSSS package and does not adversely affect any safety system or introduce any possibility of an accident of a different type than any analyzed in the FSAR, the control rod insertion limits will not be changed for Technical Specification 3.2 and the margin of safety as defined in the basis for Technical Specifications will not be reduced.

No device penetrates any pressure boundary or affects any existing piping stress analysis. No equipment shall be added to containment, so there is no effect on heat sink of containment. No cables are being added, so there is no effect on raceways and no requirements for conduits and supprot. No additional load or modification is performed to the racks so no seismic evaluation is required.

PLANT CHANGE/MODIFICATION 86-162 PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 5/15/87

SUMMARY DATE: 08-03-87

REVISION: 0

REMOVAL OF CCW PIPING TO THE PRIMARY SHIELD COOLERS

Summary:

PCM 86-162 removed the component Cooling Water Piping Valves instrumentation and associated hardward for the Primary Shield Coolers. The coolers were originally installed to maintain temperatures within the primary shield wall below 150° F; however, the design of the shield wall did not take credit for cooling by the primary shield coolers. Since the coolers were not required and were in need of maintenance it was decided to abandon them by performing the work described in this PCM.

Safety Evaluation:

PCM 86-162 is considered nuclear safety related since it removes portions of the component cooling water system associated with the primary shield coolers. This PCM does not adversely effect any other function of the component cooling water system. The primary shield coolers wern't required for either normal plant operation or post accident recovery. Therefore, implementation of this PCM did not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-126 PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 7/13/87

SUMMARY DATE: 8/03/87

REVISION: 0

ACCUMULATOR SI. TEST LINE SOLENOID VALVE REPLACEMENT

Summary:
This CPWO replaced the following solenoid valves SV-850 B, SV-850-D and SV-850-F.

Safety Evaluation:

This CPWO did not change the function of these solenoid valves. The replacement solenoid valves are qualified in accordance with IEEE 323 and IEEE 344. In addition seismic supports were added for these solenoid valves. This modification did not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-160

PC/M CLASSIFICATION: NNSR

UNIT: 3

TURNED OVER DATE: 7/13/87

SUMMARY DATE: 8/03/87

REVISION: 0

BAILEY TEMPERATURE TRANSMITTER REPLACEMENT FOR TPCW

Summary:

This CPWO replaced TE-1432 and associated transmitter TT-2201. This CPWO also added a thermowell in which the new temperature element without compromising the TPCW heat exchanger pressure boundary.

Safety Evaluation:

This CPWO did not change the function of this temperature loop. The TPCW system is non safety related and non seismic. The thermowell does not degrade the TPCW piping classification. This modification did not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-037

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 6/9/87

SUMMARY DATE: 7/25/87

REVISION: 0

ICW PUMP FOUNDATION REPAIR ANCHOR BOLT REPLACEMENT

Summary:

This CPWO provided for replacement of Anchor Bolts & pump bases for Unit 3 ICW Pump Foundations and repair of an Intake screen backwash pump basket strainer drain pipe, and ICW pump grounding cable. The drain pipe and grounding of the anchor bolts.
The replacement bolts are larger and longer to provide increased anchorage capacity for the ICW pumps. This CPWO improved the reliability of the ICW pumps.

Safety Evaluation:

This CPWO is considered safety related since it is associated with the intake cooling water pump support. This CPWO does not adversely affect any function of the ICW system. It does not increase the probability of an accident or the possibility of an accident, different than that in the FSAR.
The consequences of an accident are not changed from those described in the FSAR. The Margin of safety is not reduced from that currently defined in the Technical Specifications.

PLANT CHANGE/MODIFICATION 87-102 PC/M CLASSIFICATION: QUALITY RELATED

CPWO

UNIT: 4

TURNED OVER DATE: 06/09/87

SUMMARY DATE: 8/4/87

REVISION: 0

REACTOR VESSEL HEAD INSULATION-REFLECTIVE REPLACEMENT

Summary:

Boric acid accumulation from the Conoseal Leak within nine reflective insulation panels has reduced the thermal performance of the insulation. Cleaning of the insulation has proven ineffective and replacement was deemed prudent. This CPWO installed replacement insulation panels fabricated of all stainless steel interior and exterior plates with stainless steel foil filler. This design is considered an upgrade due to the absence of the aluminum foil filler originally used. The additional weight of the replacement panels (160 lb/panel) has been evaluated and the increased load are acceptable. The new panels are fabricated to the original dimensions, therefore the installation process is unchanged.

Safety Evaluation:

CPWO 87-102 is considered quality related since it attaches to the reactor vessel. This CPWO does not adversely affect the reactor vessel or associated components. The replacement insulation meets the criteria of specification 67449 and is therefore acceptable from a thermal performance standpoint. Therefore, implementation of this CPWO did not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-101 PC/M CLASSIFICATION: QR

UNIT: 4

TURNED OVER DATE: 6/9/87

SUMMARY DATE: 8/4/87

REVISION: 0

REACTOR VESSEL HEAD INSULATIO -PERMANENT REPLACEMENT

Summary:

As a result of damage from the leaking Unit 4 Conoseal, it was necessary to replace the "Permanent" reactor vessel insulation. The original insulation consisted of block type unibestos (or equal) with asbestos cement filler, asbestos tape coating and 1/2 " layer of "one cote" cement over tape. The replacement material consists of two 1½" layers of B & W Kaowool, Fiberfrax cloth coating with 1/4" thick coating of fiberfrax cement and waterproof coating of GE SM-2010 Silicon Release Emulsion. The replacement materials conform to the requirements or Reg. Guide 1.36 and have an equivalent or better performance as compared to the original insulation (eg. thermal performance, materials etc).

Safety Evaluation:

CPWO 87-101 is considered Quality Related since it is applied to the Reactor Vessel. This CPWO does not adversely affect the Reactor Vessel Head or appurtenances. Application or the Reactor Vessel Head Permanent insulation does not represent an unreviewed safety question.

PLANT CHANGE/MODIFICATION 87-177

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 06/11/87

SUMMARY DATE: 08/05/87

REVISION: 0

CONTAINMENT SPRAY RESTRICTING ORIFICE

Summary:

This PC/M added a restricting orifice on the Containment Spray Pump discharge flange, to reduce the spray pumps injection flow rate to an acceptable level, with respect to NPSH requirements and accident analysis assumptions. The PC/M also: affected several Emergency Operating Procedures to ensure pump cavitation is avoided, and avoid violating the Emergency Diesel Generator Loading; and a setpoint change was made to the RWST low level alarm to preclude pump cavitation.
This PC/M evaluated all POST-LOCA operating modes to determine all pump flows and required flows.

Safety Evaluation:

The probability of an occurrence of an accident or malfunction of equipment previously evaluated is not increased, since the Containment Spray System's only function is to mitigate accidents. The consequences of an accident will not change as diesel loading requirements are still acceptable.

- Due to this systems function as accident mitigation and the minimal change in system configuration, the possibility of an accident of a different type than those currently evaluated in the FSAR is not increased.

- With respect to malfunction of a different type than those previously analyzed in the FSAR, the changes in stress load, support loading, and diesel generator loading are all within the allowables and therfore preclude the increase in probability of failure or malfunction of any equipment important to safety/

- For the reasons stated above, the margin of safety as currently defined in the Technical Spcifications is not reduced due to this PC/M.

PLANT CHANGE/MODIFICATION 86-100

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 10/31/86

SUMMARY DATE: 8/17/87

REVISION: 0

NIS SOURCE RANGE PREAMP REPLACEMENT

Summary:

This CPWO replaced both Unit 4 Source Range preamplifiers (N-31 & N-32) with new Westinghouse preamplifiers. The new preamplifiers have better noise rejection characteristics. This change was part of an overall effort to reduce noise on the excore channels.

Safety Evaluation:

The source range preamplifiers are nuclear safety related because these channels are required by Technical Specifications during refueling operations and plant startup. The replacement preamplifiers are fully qualified in accordance with IEEE 323 & 344. The function of the preamplifier was not changed (this was simply a one for one part replacement). This CPWO did not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-184

PC/M CLASSIFICATION: ITS

UNIT: 4

TURNED OVER DATE: 5/30/87

SUMMARY DATE: 8/17/87

REVISION: 0

RPI INVERTER REGULATOR TRANSFORMER REPLACEMENT

Summary:

This package covers the replacement of the failed Solotron Line Voltage Regulator.

Safety Evaluation:

The replacment regulator has the same nominal voltage and KVA ratings. This change does not involve an unreviewed safety question because the probability of occurrence or the consequences of a design basis accident are not increased. The malfunction of safety related equipment previously evaluated in the FSAR is not increased. The Possibility for an accident or malfunction, of a different type than evaluated previously in the FSAR is not created. The margin of safety as defined in the bases for a Technical Specification is not reduced.

PLANT CHANGE/MODIFICATION 83-50

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 2/12/87

SUMMARY DATE: 8/17/87

REVISION: 0

CR-2

MASONARY WALL MODIFICATIONS - UNITS 3 & 4

Summary:

An inspection of masonry walls in the Auxiliary Building, Control Building, Turbine Building, Diesel Generator Building and Stream Generator Feed Pump Enclosure indicated these walls were erected without rebar and/or grout. As these walls supported safety related components, these walls were modified to be able to carry the design loads.

Safety Evaluation:

This modification improved the walls load bearing strengths by adding rebar and grout or as required rebuilding the wall. This modification does not increase the possiblity of occurrence or the consequences of an accident or, malfunction of equipment previously evaluated in the FSAR , does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR, and does not reduce the margin of safety as defined in the basis for any technical specification.

PLANT CHANGE/MODIFICATION 87-210 (CPWO) PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 6/12/87

SUMMARY DATE: 8/17/87

REVISION: 0

REPLACEMENT OF SUPPORTS 4-SIH-42 AND 4-PRWH-11

Summary:
Two Pipe Supports 4-SIH-42 and 4-PRWH-11 on the line from the Refueling Water Storage Tank to the Charging Pump Suction were replaced with new supports designed to provide the same vertical and lateral restraint. The supports were located within 10" of the old supports. The old supports were considered inadequate due to corrosion.

Safety Evaluation:

This line and supports are nuclear safety related. The design of the supports provide the same vertical and lateral support and are located at the same design location. The supports are mounted on the trench wall vice floor. The wall has been analyzed to be an adequate mounting location. This modification does not increase the possibility of occurance or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR and does not reduce the margin of safety as defined in the basis for any Technical Specification.

PLANT CHANGE/MODIFICATION 84-16

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 4/13/87

SUMMARY DATE: 8/17/87

REVISION: 0

RHR ISOLATION VALVE CIRCUIT MODIFICATION

Summary:

The change installs an override pushbutton on VPB to allow the operator to open MOV750, MOV751 should they close on a momentary high pressure spike when on RHR. These pushbuttons allow the valve to be opened before it completes the complete closure cycle thus preventing a loss of RHR letdown and subsequent challenges to the OMS System.

Safety Evaluation:

This change is nuclear safety related because it affects the RHR system isolation function. It does not present an unreviewed safety question because, it does not change RHR isolation, but only allows the valves to be opened rapidly after the isolation signal has cleared.

PLANT CHANGE/MODIFICATION 85-182

PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 10/29/86

SUMMARY DATE: 08/17/87

REVISION: 0

CHEMICAL ADDITION LINES SUPPORT REPAIRS

Summary:

The chemical addition lines to the main feedwater piping were found to have inadequate supports for seismic Category 1. This modification provided new supports, new anchors and modified existing supports to adequately support the chemical addition line.

Safety Evaluation:

This modification improved the line supports to meet Seismic Category 1 criteria. The line is nuclear safety related. This modification does not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR, does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR and does not reduce the margin of safety as defined in the basis for any Technical Specification.

PLANT CHANGE/MODIFICATION 85-139

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 09/05/86

SUMMARY DATE: 08/17/87

REVISION: 0

REMOVAL OF VALVES 4-524, 4-525 AND ASSOCIATED PIPING (PRESSURIZER MINI-SPRAY)

Summary:

On Turkey Point Unit 4, pressurizer spray mini-bypass lines and valves exist (Ref. PC/M 82-161) for the two relocated spray valves (with new bodies). Additionally, two mini-bypass lines and valves use to exist for the abandoned original plant spray valve bodies. The two original installation bypass lines which are no longer in service were a high source of radiation. Their function is now performed by the newer valves installed under PC/M 82-161.

This PC/M removed the original 3/4" mini-spray bypass lines and valves provided for the valve bodies abandoned per PC/M 82-161. These mini-spray bypass lines were located above the Containment 54' elevation and were a high source of radiation. Their removal greatly improved ALARA concerns in the immediate vicinity and eliminated the need to shield these valves and piping.

This removal did not affect the capability of the new mini-spray bypass lines (currently at 14' elevation).

Safety Evaluation:

This design removed equipment whose function is no longer required. The removal was to further limit radiation to amounts as low as reasonably achievable. The equipment that superseded the function of the equipment removed by this PC/M received the appropriate level of safety analysis via PC/M 82-161. The removal of equipment in this PC/M has undergone sufficient review to assure that all current analysis (stress analysis or the 4" pipe) or design practices (provision of sufficient minispray flow) impacted by the equipment has not been detrimentally affected by its removal

For the above reasons, the probability or consequences of an accident previously evaluated in the FSAR have not increased. Furthermore, the probability or consequence of malfunction of equipment important to safety remain unchanged, and the possibility of an accident of a different type than any previously analyzed has not been created. Finally, the margin of safety as defined in the Technical Specifications is not affected by this equipment removal.

PLANT CHANGE/MODIFICATION 85-181

PC/M CLASSIFICATION: NSR

UNIT: 4

TURNED OVER DATE: 10-3-86

SUMMARY DATE: 8/17/87

REVISION: 0

REMOVAL OF INACTIVE NITROGEN BLANKET LINE

Summary:

This PCM removed an existing plant nitrogen system supply line connected to the Train 1 Auxiliary Feedwater Piping of Unit 4. This line was used during construction for blanketing portions of the secondary side of the plant with nitrogen. The line was no longer in service.

Safety Evaluation:

This modification is consistent with all applicable design requirements for the Auxiliary Feedwater System and affects no other systems. The QC requirements imposed are sufficient to ensure that the modification is made accurately and adequately.

Therefore, this modification does not involve an unreviewed safety question and is considered acceptable.

PLANT CHANGE/MODIFICATION 84-11

PC/M CLASSIFICATION: NSR

UNIT: 3

TURNED OVER DATE: 7/3/85

SUMMARY DATE: 8/17/87

REVISION: 0

MODIFICATION TO PRESSURIZER SPRAY SYSTEM - FUNCTIONALITY (I.C.)

Summary:

All work in this PC/M was completed under Change Request #4 to PC/M 81-146. This PC/M was for the transfer of documentation only in order to clarify the Engineer of Record for PC/M 81-146. The work was support modifications to the Pressurizer Spray System Functionality (Inside Containment) required for compliance with NRC IE Bulletin 79-14.

Safety Evaluation:

The modifications and analyses ensured that the design criteria of the original piping system design documents were met. Therefore, the probability or consequences of an accident previously evaluated in the FSAR have not increased. Futhermore, the probability or consequence of malfunction of equipment important to safety remain unchanged, and the possibility of an accident of a different type than any previously analyzed has not been created. Finally, the margin of safety as defined in the Technical Specifications was not affected by this PC/M.

PLANT CHANGE/MODIFICATION 85-141 PC/M CLASSIFICATION: OR

UNIT: 3

TURNED OVER DATE: 6/29/87

SUMMARY DATE: 8/17/87

REVISION: 0

FUEL TRANSFER SYSTEM MANIPULATOR CRANE DUAL CABLE MODIFICATION

Summary:

This PC/M covers modifications to the Unit 3 manipulator crane. These modifications consist of:

1. Upgrading the manipulator crane from a single cable hoist to a dual cable hoist.
2. The installation of a new hoist load indicator system.
3. The installation of a load test fixture used to load test the main hoist on the manipulator crane. (An existing structure will be used to load test the manipulator crane's auxiliary hoist.)

This PC/M added redundancy to the cable hoist portion of the refueling machine. This enhancement increased reliability, will reduce maintenance time, and incorporated the latest state of the art capabilities into the refueling system.

Safety Evaluation:

The installation of the manipulator crane modifications does not have any direct effect on nuclear safety since the manipulator crane is not safety related. This modification could indirectly affect nuclear safety since a crane accident could result in a design basis accident. However, the modifications associated with this PCM include changes which reduce the possibility of potential accidents.

An unreviewed safety question, as defined by 10CFR 50.59 does not exist and no Technical Specifications are affected.

PLANT CHANGE/MODIFICATION 86-121

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 5/30/87

SUMMARY DATE: 8/17/87

REVISION: 0

CONTROL POWER FUSE REPLACEMENT

Summary:

This Package covers the replacement of Gould Shawmut Control Power Fuse Type A2Y 3 AMP Type 2 with Type A6Y 3 AMP Type 2.

Safety Evaluation:

This change does not involve an unreviewed safety question because the probability of occurrence or the consequences of a design basis accident are not increased. The malfunction of safety related equipment previously evaluated in the FSAR is not increased. The possibility for an accident or malfunction of a different type than evaluated previously in the FSAR is not created. Also the margin of safety as defined in the basis for a Technical Specification is not reduced. The preceeding is correct because all electrical specifications of the replacement fuse are identical to the old fuse. The voltage rating of the replacement fuse is higher than the old fuse.

PLANT CHANGE/MODIFICATION 86-107

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 5/30/87

SUMMARY DATE: 8/17/87

REVISION: 0

MCC CONTROL POWER FUSE REPLACEMENT

Summary:

This package covers the replacement of Gould Shawmut Control Power fuse Type A2Y 3 AMP Type 2 with Type A6Y 3 AMP Type 2. This control Power Fuse is used on the Emergency Containment Filter 3B

Safety Evaluation:

This change does not involve an unreviewed safety question because the probability of occurrence or the consequence of a design basis accident are not increased. The malfunction of safety related equipment, previously evlauated in the FSAR is not increased. The possibility for an accident or malfunction of a different type than evaluated previously in the FSAR is not created. The margin of safety as defined in the basis for a Technical Specification is not reduced. The preceeding is correct because the replacement is equal or better when compared to the old fuse in all categories.

PLANT CHANGE/MODIFICATION 86-026

PC/M CLASSIFICATION: SR

UNIT: 4

TURNED OVER DATE: 5/30/87

SUMMARY DATE: 8/17/87

REVISION: 0

4KV SWITCHGEAR BREAKER ELEVATING MECHANISM REBUILDING

Summary:

This CPWO involves the rebuilding and reinstallation of the "E" shaped right and left side elevating mechanisms on the 4KV Switchgear Startup Transformer Breaker 4AA05; and the Unit Auxiliary Transformer Breaker 4AA02.

Safety Evaluation:

The newly installed parts do not perform any load bearing function except during elevating the breaker assembly. Since the rebuilding of this elevating mechanism does not affect the seismic capabilities of this "E" shaped bracket to support the breaker assembly there will be no detrimental effect upon the, capabilities of this part to perform its design function. Therefore it can be said that: This change does not involve an unreviewed safety question because the probability of occurrence or the consequences of a design basis accident are not increased. The malfunction of safety related equipment previously evaluated in the FSAR is not increased. The possibility for an accident or malfunction of a different type than evaluated previously in the FSAR is not created. The margin of safety as defined in the bases for a Technical Specification is not reduced.

PLANT CHANGE/MODIFICATION 85-11

PC/M CLASSIFICATION: SR

UNIT: 3

TURNED OVER DATE: 6/3/87

SUMMARY DATE: 8/17/87

REVISION: 0

MODIFICATION OF 4160 V BREAKER HH SWITCHES

Summary:

This PC/M changed the 4160 V Bkr. Position Switches "52 HH" for a better designed switch and added wiring so that the breaker position white indications light monitors the action of the "52HH" switches.

Safety Evaluation:

This PC/M is safety related. The new switches and wiring enhance the 4160 V Bkr. operation. The probability of occurrence of an accident previously evaluated in the FSAR is not increased, and an unreviewed safety question is not involved.

PLANT CHANGE/MODIFICATION 85-149	PC/M CLASSIFICATION: QR
	UNIT: 3
	TURNED OVER DATE: 5-7-87
	SUMMARY DATE: 6-18-87
	REVISION: 0

SFP AIR INLET DAMPER REPLACEMENT

Summary:

This PC/M replaced the Unit 3 Spent Fuel Pit Room Fresh Air Inlet Dampers.

Safety Evaluation:

This PC/M is quality related in accordance with FSAR Appendix 5A requirements with no unreviewed safety question since the probability/consequences of an accident previously evaluated in the FSAR has not increased, nor was the possibility of an equipment malfunction/accident important to safety previously evaluated in the FSAR. This modification will not decrease the margin of safety as defined in the bases of any Technical Specification.

ii) PROCEDURE CHANGES

The following procedures were changed, reviewed, and approved and reissued during the reporting period. The procedure changes are as described below and only those procedure changes constituting changes in the procedures as described in the Final Safety Analysis Report (FSAR) are reported. Minor or routine procedure changes not affecting procedures as described in the FSAR are not reported.

1) A review of the manufacturer's recommendations for cooling water temperature for the diesel driven fire pump determined that the criteria in the procedure did not match the manufacturer's recommendations. The following procedure was changed to reflect the manufacturer's recommendations:

   a) OP 15524: Fire Protection Pumps and Power Supplies - Periodic Test

   Safety Evaluation Summary:
   The proposed change only changes cooling water temperature limits to those specified by the pump manufacturer. This change does not alter the pump operation or change the pumps capability to perform its intended function as specified by the manufacturer. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

   Date of Change: July 3, 1986

2) The existing procedure for fire stop and cable tray fireproofing was revised to identify specific fire areas that are required to have cables coated, and the criteria for applying and maintaining the coatings in compliance with Regulatory requirements. In addition various portions were removed that were no longer applicable. The following procedure was revised:

   a) MP 0725: Fire Stop and Cable Tray Fireproofing (New title is Application and Maintenance of Flame Retardant Cable Coatings (Flamemastic 77))

   Safety Evaluation Summary:
   The evaluation serves to clarify the commitments that have been made in regards to the use of cable coatings or IEEE-383 qualified cable. As such, the safety of the units will not be reduced if qualified or coated cables are provided in only the areas specified by detailed fire hazard analysis reports. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

   Date of Change: July 3, 1986

3) Special Tests were performed to determine the proper component cooling water system flow balance for Unit 4. The results of these tests were used to revise the following procedures:

a)	4-OP-030	Component Cooling Water System
b)	0-ADM-205	Administrative Control of Valves, Locks, and Switches
c)	AP 0103.19	Monthly Verification of Safety Related Systems Flowpaths
d)	AP 0103.32	Reactor Cold Shutdown Conditions

Safety Evaluation Summary

The system alignment established by the special test will serve to ensure that the system is capable of satisfying design basis heat removal requirements during accident conditions. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: July 24, 1986

4) A study of emergency diesel generator (EDG) loading determined that a potential existed to exceed the FSAR and emergency operating procedure loading limits for the EDGs. Based on this plant changes were made along with revising the following procedures:

a)	3(4)-EOP-ES-0.0	Rediagnosis
b)	3(4)-EOP-ES-1.3	Transfer to Cold Leg Recirculation
c)	3(4)-EOP-ES-1.4	Transfer to Hot Leg Recirculation
d)	3(4)-EOP-ES-3.1	Post-SGTR Cooldown Using Backfill
e)	3(4)-EOP-ES-3.2	Post-SGTR Cooldown Using Blowdown
f)	3(4)-EOP-ES-3.3	Post-SGTR Cooldown Using Steam Dump
g)	3(4)-EOP-ECA-0.0	Loss of All AC Power
h)	3(4)-EOP-ECA-3.2	SGTR With Loss of Reactor Coolant-Saturated Recovery Recovery Desired
i)	3(4)-EOP-ECA-3.3	SGTR Without Pressurizer Pressure Control
j)	3(4)-EOP-F-0	Critical Safety Function Status Trees
k)	OP-057	Containment Normal Ventilation and Cooling System
l)	OP-0205.2	Reactor Shutdown - Hot Standby to Cold Shutdown Condition
m)	OP-10304.6	Computer Room Chilled Water System - Operating Instructions
n)	H:NPO-3.3	Nuclear Plant Operator
o)	H:NPO-4.3	Nuclear Plant Operator

Safety Evaluation Summary

Implementation of the EDG loading evaluation assumptions demonstrates that the FSAR load evaluation remains bounding, and that the EDGs meet the loading requirements set forth in the PTPN Technical Specifications. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: July 27, 1987.

5) A study of emergency diesel generator (EDG) loading determined that a potential existed to exceed the FSAR and emergency operating procedure loading limits for the EDGs. Based on this plant changes were made along with revising the following procedures:

a)	3(4)-EOP-E-0	Reactor Trip or Safety Injection
b)	3(4)-EOP-ES-0.1	Reactor Trip Response
c)	3(4)-EOP-ES-0.2	Natural Circulation Cooldown
d)	3(4)-EOP-ES-0.3	Natural Circulation Cooldown With Steam Void in Vessel (With RVLMS (QSPDS))
e)	3(4)-EOP-ES-0.4	Natural Circulation Cooldown With Steam Void in Vessel (Without RVLMS)
f)	3(4)-EOP-E-1	Loss of Reactor or Secondary Coolant
g)	3(4)-EOP-ES-1.1	SI Termination
h)	3(4)-EOP-ES-1.2	Post LOCA Cooldown and Depressurization
i)	3(4)-EOP-E-3	Steam Generator Tube Rupture
j)	3(4)-EOP-ECA-0.1	Loss of All AC Power Recovery Without SI Required
k)	3(4)-EOP-ECA-0.2	Loss of All AC Power Recovery With SI Required
l)	3(4)-EOP-ECA-2.1	Uncontrolled Depressurization of All Steam Generators
m)	3(4)-EOP-ECA-3.1	SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired
n)	3(4)-ONOP-004	Loss of Offsite Power
o)	3(4)-OP-006	480 Volt Switchgear System
p)	OP 9104.1	Main Transformer - Periodic Tests
q)	ONOP 9108.1	Main Transformer - Malfunction
r)	3(4)-OP-007	480 Volt Motor Control Centers

Safety Evaluation Summary

Implementation of the EDG loading evaluation assumptions demonstrates that the FSAR load evaluation remains bounding, and that the EDGs meet the loading requirements set forth in the PTPN Technical Specifications. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: July 28, 1987.

6) During a review of the 4160 volt bus lockout schemes for both units, two types of devices were identified, whose failure could adversely affect the ability of safety related equipment to perform their intended safety function. Based on this the following procedures were revised:

a)	3(4)-EOP-ES-0.1	Reactor Trip Response
b)	3(4)-EOP-E-0	Reactor Trip or Safety Injection

Safety Evaluation Summary:

The addition of cautions on ensuring that the battery chargers are energized within 30 minutes will provide additional assurance that the requirements of the FSAR are met. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: August 15, 1986

7) A temporary procedure was written to provide instructions for running the B AFW pump to maintain the Unit 3 steam generator levels, so that maintenance can be performed on PI-3-1435. The following temporary procedure was developed:

a)	TP 277	Use of AFW to Maintain SG Levels During Repair of PI-3-1435

Safety Evaluation Summary:
The TP allows operation of train 2 AFW to maintain plant power at equal to or less than 5 percent in accordance with the AFW system design. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: August 24, 1986

8) The FSAR states that the mechnical overspeed trip setpoint has a value of 11% above rated speed (1998 RPM). The following procedure was revised with an on-the spot change to provide an acceptance band for this setpoint.

a)	OP-8004.1	Turbine Generator - Overspeed Trip Test

Safety Evaluation Summary
Revising the overspeed setpoint to less that 11% of rated speed is a more conservative setting with regard to protection of the turbine. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: September 6, 1986

9) In order to incorporate the requirements of Technical Specification Amendment 118 (Facility Operating License DPR-31 for Unit 3) and Amendment 112 (Facility Operating License DPR-41 for Unit 4) which incorporated a new Technical Specification for the Standby Steam Generator Feedwater System, the following procedures were revised:

a)	OP-0202.1	Reactor Startup - Cold Shutdown to Hot Standby Conditions

Date of Change: July 17, 1986

b)	AP-0190.16	Scheduling and Surveillance of Periodic Tests and Checks

Required by Technical Specifications

Date of Change: September 26, 1986

c)	AP-0103.7	Reports Required by Technical Specifications and 10 CFR
d)	ONOP-0208.9	Annunciator List - Panel G - Miscellaneous
e)	OP-16001.2	Technical Specification Surveillance Requirements for Core Refueling
f)	0-OP-018	Demineralized Water System
g)	0-OP-074.1	Standby Steam Generator Feedwater System
h)	0-ONOP-074.1	Standby Steam Generator Feedwater System Operation With Loss of Offsite Power and Loss of Auxiliary Feedwater
i)	0-OSP-200.1	Schedule of Plant Check and Surveillances
j)	0-OSP-074.3	Standby Steam Generator Feedwater Pumps Availability Test
k)	0-OSP-074.4	Standby Steam Generator Feedwater Pumps/Cranking Diesels Test
l)	NPO-4:2	Nuclear Plant Operator Logsheet

Date of Changes: October 2, 1986

m)	AP 0103.12	Notification or Significant Events to NRC

Date of Change: October 9, 1986

10) Procedure changes were made to incorporate the operation of diesel air compressors as described in temporary procedures (TP) 250 and 251 into existing plant procedures. These diesel air compressors were installed for the emergency diesel generator loading evaluation. The following procedures were changed.

a)	3(4)-OP-013	Instrument Air System
b)	0-ONOP-013	Loss of Instrument Air

Safety Evaluation Summary
Implementation of the EDG loading evaluation assumptions demonstrates that the FSAR load evaluation remains bounding, and that the EDGs meet the loading requirements set forth in the PTPN Technical Specifications. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: July 28, 1987.

11) Due to a change in a breaker number, the following procedures were revised:

a)	3(4)-OP-005	4160 Volt Buses A and B
b)	0-ONOP-074.1	Standby Feedwater System Operation With Loss of Offsite Power and Loss of Auxiliary Feedwater
c)	ONOP-0208.8	Annunciator List - Panel F - Electrical
d)	ONOP-0208.16	Annunciator List - Panel J - Auxiliary Electrical Power
e)	ONOP 9408.2	Energizing 4KV Buses Using the Cranking Diesels Bus Tie Lines or Startup Transformer from the Opposite Unit
f)	ONOP-9308.1	Startup Transformer - Malfunction

g) ONOP-9308.2 "C" Bus Transformer - Malfunctic

Safety Evaluation Summary:
The changes consisted of administrative changes only and did not affect the operation of the breaker. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: October 7, 1986

12) The FSAR states that the mechnical overspeed trip setpoint has a value of 11% above rated speed (1998 RPM). The following procedure was revised to permanently incorporate a prior on-the-spot-change approved on September 6, 1986.

a) OP-8004.1 Turbine Generator - Overspeed Trip Test

Safety Evaluation Summary
Revising the overspeed setpoint to less that 11% of rated speed is a more conservative setting with regard to protection of the turbine. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: October 14, 1986

13) In order to load test the 4A battery in the vital DC system a temporary procedure was written to perform this evolution. The following temporary procedure was written:

a) TP 286 Unit 4 - Removal and Return to Service 4A DC Battery for Load Test

Safety Evaluation Summary
The evaluation concludes that during the test the 3B and 4A DC busses will be interconnected to keep all four instrument channels functional. The plant will be operating under a DC system LCO per Technical Specification 3.7.2 during the plant condition, while battery 3B and battery chargers 3B and 4S feed both DC busses 3B and 4A. Neither the basic design nor the permanent configuration of any equipment are affected by this change. The interconnection of DC busses 3B and 4A will be governed by the requirements of Technical Specifications for a battery out of service. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: November 25, 1986

14) In order to load test the 3A battery in the vital DC system a temporary procedure was written to perform this evolution. The following temporary procedure was written:

a) TP 290	Unit 3 - Removal and Return to Service 3A DC Battery for Load Test

Safety Evaluation Summary
The evaluation concludes that the temporary battery can be used to replace the 3A battery to feed bus 3A with a DC system LCO invoked during the service testing of the 3A battery. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: November 28, 1986

15) A review of the design of the control room ventilation system discovered a the potential for loss of the control room ventilation system that could prevent the system from performing its intended function. Based on this the following procedures were developed:

a) TP 291	Loss of Control Room Ventilation System (CRVS) Air Conditioning

Safety Evaluation Summary
The interim actions taken ensure that sufficient control room cooling and atmosphere cleanup capability exists for accident scenarios described in the FSAR. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: December 18, 1986

16) In order to incorporate the requirements of Technical Specification Amendment 120 (Facility Operating License DPR-31 for Unit 3) and Amendment 114 (Facility Operating License DPR-41 for Unit 4) which incorporated a revision to the emergency diesel generator 18 month preventative maintenance requirements which changed them to coincide with the Unit 3 refueling outage schedule. The following procedures were revised:

a) OP-4304.3	Emergency Diesel Generator - Eight Hour Full Load Test and Load Rejection
b) OP 16001.2	Technical Specification Surveillance Requirements For Core Refueling
c) 0-OSP-200.1	Schedule of Plant Checks and Surveillances
d) 0-OSP-200.2	Plant Startup Surveillances
e) 3(4)-OSP-203	Engineered Safeguards Integrated Test

Date of Changes: January 7, 1987

17) A review of the design of the control room ventilation system discovered a the potential for loss of the control room ventilation system that could prevent the system from performing its intended function. This evaluation was revised to allow operation of the heating feature of the control room heating/ air conditioning system. Based on this the following procedure was revised:

a) TP 291 — Loss of Control Room Ventilation System (CRVS) Air Conditioning

Safety Evaluation Summary

The interim actions taken ensure that sufficient control room cooling and atmosphere cleanup capability exists for accident scenarios described in the FSAR. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: January 29, 1987

18) In order to load test the 4B battery in the vital DC system a temporary procedure was written to perform this evolution. The following temporary procedure was written:

a) TP 310 — Units 3 and 4 - Removal and Return to Service 4B DC Battery for Load Test

Safety Evaluation Summary

The evaluation concludes that the temporary battery can be used to replace the 4B battery to feed bus 4B with a DC system LCO invoked during the service testing of the 3A battery. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: March 10, 1987

19) In order to lower the Unit 3 spent fuel pit level for maintenance activities, the following temporary procedure was developed:

a) TP 313 — Unit 3 SFP Level Reduction for Maintenance

Safety Evaluation Summary:

During the reduction of SFP level, no fuel movement shall occur, no crane operation with loads shall be conducted, and the temperature of the SFP water shall not be allowed to exceed 180 degrees Fahrenheit. By following these restrictions, the proposed temporary procedure will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: March 17, 1987

20) A temporary procedure was written to provide instructions for the recirculation and draining of Demineralized Water Storage Tank. This was required after routine samples of the tank contained very low levels of cobalt 58 and cobalt 60. The tank was isolated and analysis of the tank indicated that the concentrations were acceptable for direct release to the atmosphere. These instructions include all necessary prerequisites and guidelines for the tank to be recirculated for sampling requirements and for the gravity draining to the discharge canal. The procedure also ensures that the tank is isolated from external sources of liquids during the draining process and during non-recirculation modes of operation.

a) TP 334	DWST Recirculation and Release to Discharge Canal

Safety Evaluation Summary:
The flushing and discharge of the contents of the DWST to the cooling canal neither involves nor is associated with any accident scenario described in the FSAR. The TP incorporates the requirements of Technical Specification 3.9 for radioactive liquid releases. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: April 27, 1987

21) In order to incorporate the requirements of Technical Specification Amendment 123 (Facility Operating License DPR-31 for Unit 3) and Amendment 116 (Facility Operating License DPR-41 for Unit 4) which incorporated a revision to the reporting requirements of section 6.0. The following procedures were revised.

a) AP-0103.7	Reports Required By Technical Specifications and 10 CFR
b) ONOP-2608.2	Chemical and Volume Control System Malfunction of Boron Concentration Control System
c) MP-15537.3	Surveillance of Penetration Fire Barriers
d) ONOP-15538	Fire and Smoke Detection System and Fixed Fire Protection Equipment/Systems - Operating Instructions

Date of Changes: May 5, 1987

22) An evaluation of the switchover to cold leg recirculation during a LOCA determined that a 10 minute interruption of flow could not be supported using the latest analytical assumptions. The following procedures were revised.

a) 3(4)-EOP-ES-1.3	Transfer to Cold Leg Recirculation
b) 3(4)-EOP-ES-1.4	Transfer to Hot Leg Recirculation

Safety Evaluation Summary
The revisions have been made to maintain continuous SI flow to the RCS following

large breaks (unless spray cooling for the emergency containment filters is required or where RHR flow indicator FI-605 exhibits erratic behavior) and to minimize the time of interruption of SI flow during switchover to recirculation for small breaks. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: May 26, 1987

23) Due to problems with nitrogen inleakage through the boric acid transfer pumps, procedure changes were made to have the nitrogen supply normally isolated to the pumps. The following procedures were revised:

a)	ONOP-2608.2	Chemical and Volume Control System Malfunction of Boron Concentration Control System
b)	0-OP-065.3	Nitrogen Gas Supply System
c)	0-OSP-201.2	SNPO Daily Logs
d)	0-ADM-205	Administrative Control of Valves, Locks and Switches

Safety Evaluation Summary

These changes do not introduce any conditions that are beyond the design capabilities of the boric acid transfer pumps or represent a condition that is outside the design basis for the boric acid transfer pumps. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: June 10, 1987

24) In order to incorporate the requirements of Technical Specification Amendment 124 (Facility Operating License DPR-31 for Unit 3) and Amendment 118 (Facility Operating License DPR-41 for Unit 4) which incorporated a revision to the requirements for the auxiliary feedwater system and the condensate storage tanks. These changes include adding the modes of applicability and additional action statements. The following procedures were revised as a result of this amendment.

a)	3(4)-OP-018.1	Condensate Storage Tank (CST)
b)	3(4)-OP-075	Auxiliary Feedwater System
c)	3(4)-OSP-075.1	Auxiliary Feedwater Train 1 Operability Verification
d)	3(4)-OSP-075.2	Auxiliary Feedwater Train 2 Operability Verification
e)	3(4)-OSP-075.3	AFW Nitrogen Backup System Low Pressure Alarm Setpoint and Leakrate Verification
f)	3(4)-OSP-075.6	Auxiliary Feedwater Train 1 Inservice Test
g)	3(4)-OSP-075.7	Auxiliary Feedwater Train 2 Inservice Test
h)	0-OSP-075.9	AFW Overspeed Test

Date of Changes: June 18, 1987

i)	3(4)-OSP-201.1	RCO Daily Logs

Date of Changes: July 1, 1987

25) An evaluation of the HVAC system for the DC equipment/inverter rooms discovered scenarios which could result in a loss of the HVAC function. The loss of the HVAC could result in elevated temperatures in the affected rooms and could affect the operability of equipment in these rooms. The following procedures were developed or revised.

a)	3(4)-ONOP-004	Loss of Offsite Power
b)	TP-347	DC Equipment and Inverter Rooms Supplemental Cooling Monitoring and Standby Conditions
c)	TP-348	DC Equipment and Inverter Rooms Continuous Supplemental Cooling
d)	TP-349	DC Equipment and Inverter Rooms Supplemental Cooling During Normal Conditions
e)	TP-350	DC Equipment and Inverter Rooms Supplemental Cooling During Fire Conditions
f)	TP-351	DC Equipment and Inverter Rooms Supplemental Cooling During LOOP Conditions
g)	TP-352	DC Equipment and Inverter Rooms Supplemental Cooling During LOOP Conditions with LOCA

Date of Review: June 5, 1987

h)	TP-346	DP-412A Transfer Switch Inspection
i)	TP-355	DC Equipment/Inverter Rooms HVAC Maintenance

Date of Review: June 16, 1987

j)	TP-356	CRDM M/G Set 4A/4B Load Sharing
k)	TP-357	CRDM M/G Set 3A/3B Load Sharing

Date of Review: June 18, 1987

<u>Safety Evaluation Summary</u>
The implementation of the interim actions will ensure that temperatures remain within electrical equipment design temperatures. Therefore, the proposed procedural changes will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Changes: June 5, 1987

26) In order to inspect cable splices in the electrical penetrations, installed fire protection material (Flamemastic) was removed from selected cables for Unit 4. An evaluation was written to support the temporary removal of the Flamemastic from the affected cable trays in the Unit 4 containment penetration areas until the next refueling outage when the Flamemastic will be reapplied. Based on this evaluation, the following procedure was revised.

a) ONOP-15538 Fire and Smoke Detection System and Fixed Fire Protection

Safety Evaluation Summary

Removal of Flamemastic does not increase the probability of occurrence of a malfunctin of equipment important to safety since Flamemastic is considered a passive fire protection feature. With the Flamemastic removed, compensatory measures as incorporated in the above procedure change will be implemented. The implementation of these compensatory easures together withthe existing available fire protection features provides a high degree of assurance that at least one train of redundant safe shutdown cables will remain free of fire damage. Therefore, the proposed procedural change will not increase the probability or consequences of an accident analyzed in the FSAR nor, will it impact the functioning of any safety related equipment important to safety. No new accident or malfunction of a different type will be created and no margin of safety as defined in the basis for any Technical Specification is decreased.

Date of Change: June 5, 1987

(iii) TESTS AND EXPERIMENTS

This section contains the results and conclusions for special tests that were completed during the reporting period. Special tests still in progress at the end of the reporting period are also described.

SPECIAL TEST NUMBER	TITLE	UNIT(S)
86-20	"B" Auxiliary Feedwater Pump - Performance Test	3 and 4
86-24	Main Feedwater Check Valve Cycling Test	3 and 4
86-26	Containment Spray Recirculation Piping Flushing and Pressure Drop	4
86-28	Unit 3 TPCW H_x Performance Test to Evaluate Effectiveness of Chemical Dispersion Technique	3
86-29	Unit 4 TPCW H_x Performance Test to Evaluate Effectiveness of Chemical Dispersion Techniques	4
86-30	Condensor Cathodic Protection Special Test - Reduce Cathodic Protection Contributions to Hydriding	3 and 4
86-31	Waste Gas System Leak Test	3 and 4
86-32	Unit 3 VCT Purge Special Test	3
86-35	Unit 4 VCT Purge Special Test	4
87-01	System Generator Blowdown Control Valves Delayed Actuation Verification Test	3 and 4
87-03	WTP, DWST, and Condensate Dissolved Oxygen Monitoring Program	4
87-08	Condensor Tubing Cathodic Protection Monitoring System	3
87-11	Safety Injection System MOV Differential Pressure Stroke Testing Valves: MOV-3-843 A/B and MOV-878 A/B	3 and 4
87-13	Condensor Tubing Strain Measurement Test	3

SPECIAL TEST 86-20

"B" AUXILIARY FEEDWATER PUMP - PERFORMANCE TEST

Performed: August 7, 1986

BACKGROUND INFORMATION:

This special test was conducted to study the performance of the "B" Auxiliary Feedwater Pump after the installation of the underfiled pump impellor.

TEST RESULTS:

The study concluded that the under-filed impellor replacement improved the pump perfomance to above the design requirement.

SAFETY EVALUATION:

In Section 9.7 of this Special Test, the "B" AFW pump recirculation line back to the Condensate Storage Tank will be closed. The procedure alerts the Reactor Control Operator to maintain flow through the pump at all times during this portion of the test to prevent the pump from dead heading. In addition, the Turbine Operator shall be standing by the recirculation valve 20-277 to open it in the event of an AFW AUTO START signal. These precautions will ensure the safe operation of the AFW pump.

Sections 9.9 and 9.10 of this Special Test require the Turbine Operator to use the governer overspeed trip device to increase AFW turbine speed to 6000 ± 15 rpm and 6100 ± 15 rpm respectively. The electrical overspeed trip at 6200 ± 50 rpm and the mechanical overspeed trip at 6500 ± 50 rpm are both operable and will provide turbine protection if required.

Section 9.11 obtains data to evaluate what effect the Turbine Lube Oil Cooler line coming from the AFW pump second stage has on pump performance. The valve manipulation in this section shall be independently verified to ensure that adequate cooling is maintained to the pump and turbine lube oil and that no service water recirculates back to the Condensate Storage Tank.

This Special Test does not involve an unreviewed safety question, nor does it increase the probability or severity of an accident either previously or not previously analyzed in the FSAR. It does not increase the probability of an active or passive failure of any AFW equipment important to the function of the AFW System. Similarly, it does not increase the severity of the consequences of an AFW equipment malfunction.

SPECIAL TEST 86-24
MAIN FEEDWATER CHECK VALVE
CYCLING TEST

BACKGROUND INFORMATION:

This information is a special test to determine if the check valve shows any indication of binding of the rockshaft when rotated.

TEST RESULTS:

Test was run 7/21/86 through 7/23/86 on Units 3 & 4. Data sent to Site Engineering August 22, 1986.

SAFETY EVALUATION:

SPECIAL TEST OF MAIN FEEDWATER CHECK VALVES

SAFETY ANALYSIS:

The special test has been evaluated with the following results:

1) The probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety previously evaluated in the FSAR has not been increased due to both units being in cold shutdown and the Tech. Spec. 3.4 requirement for two coolant loops being maintained.

2) The possibility of an accident or malfunction of a different type than evaluated previously in the FSAR has not been created as all Tech. Spec. requirements for cold shutdown are being maintained. Realignment of the feedwater system does not impact the unit as feedwater is not required and typically isolated at cold shutdown with RHR in service.

3) The margin of safety as defined in the basis for a Technical Specification has not been reduced as the Tech. Spec. 3.4 requirement for two coolant loops is being maintained by one RHR pump and one RCP/SG pair availabe for service. There is no requirement that makeup be provided to the steam generators during cold shutdown; however, the test provides an extra margin of safety by providing an available makeup supply path during the test.

Therefore, the proposed special test does not involve an unreviewed safety question.

EK/rlh/:026

SPECIAL TEST 86-26

CONTAINMENT SPRAY RECIRCULATION PIPING FLUSHING & PRESSURE DROP

Background Information:

This special test was performed in order to determine if there was any blockage in the containment spray recirculation piping for unit 4. This test was performed with both unit 3 and 4 in cold shutdown.

Test Results:

This test was performed and the results were forwarded to engineering and the flow rates calculated from the pressure drop data matched design flow rates.

Safety Evaluation:

This test was written to be performed in modes 5 or 6 only. These modes do not require any containment spray trains to be operable. Therefore, special test 86-26 did not pose an unreviewed safety question.

SPECIAL TEST 86-28

Unit 3 TPCW HX Performance Test to Evaluate Effectiveness of Chemical Dispersion Technique

Background Information:

The special test was performed to give an indication of the effectiveness of adding a Calgon water treatment chemical to reduce HX tube fowling rates. This Test was to compare to Unit 3 data. Completed 10-21-86

Test Results:

The tests were inconclusive. Future evaluations will be performed.

Safety Evaluation:

The test was performed with adequate safeguards to assure that the ICW system functioned as normal. No changes were made which would adversely affect the systems operating capability. Normal line-up was maintained. Based on the above Special Test 86-28 did not pose an unreviewed safety question.

SPECIAL TEST 86-29

Unit 4 TPCW HX Performance Test to Evaluate Effectiveness of Chemical Dispersion Technique

Background Information:

The special test was performed to give an indication of the effectiveness of adding a Calgon water treatment chemical to reduce HX tube fowling rates. This Test was to compare to Unit 3 data. Completed 10-21-86

Test Results:

The tests were inconclusive. Future evaluations will be performed.

Safety Evaluation:

The test was performed with adequate safeguards to assure that the ICW system functioned as normal. No changes were made which would adversely affect the systems operating capability. Normal line-up was maintained. Based on the above Special Test 86-29 did not pose an unreviewed safety question.

SPECIAL TEST 86-30
SUMMARY

Background Information:
The purpose of this test is to reduce/eliminate the excessive hydriding of the titanium tubes in the condensers of Unit 3 and 4.

The test involves disconnecting some of the anodes in the condensers and reducing the rectifier outputs.

Test Results:
On Going

Safety Evaluation:
The configuration for this test does not involve an unreviewed safety question nor does it increase the probability of an accident because:

1. The Condenser Cathodic Protection System is non-safety related. The design intent of the Cathodic Protection System remains the same.

   This modification does not alter the operation or function of any other plant safety or non-safety systems. Therefore, the probability of malfunction of equipment important to safety has not been increased, and the consequences of malfunction of equipment to safety previously evaluated in the FSAR are not affected.

2. This test does not interface with any safety equipment. This modification does not alter the operation or function of any safety related systems. Hence, the possibility of an accident of a different type than analyzed in the FSAR is not increased, and the possibility of a malfunction of a different type than analyzed in the FSAR has not been created.

3. This modification does not affect the integrity, operation or function of any safety related system addressed in the Technical Specifications. Therefore, the margin of safety as defined in the basis for any plant specification is not decreased.

In conclusion, the subject special test does not constitute an unreviewed safety question.

SPECIAL TEST 86-31

WASTE GAS SYSTEM LEAK CHECK

Background Information:

Subject test was completed on 12/86

Test Results:

The Test consisted of isolating the equipment that uses the Waste Gas System: CVCS, Hold-up Tanks, Spent Resen Storage Tank, Reactor Coolant Drain Tanks, Sample Rooms, Gas Stripped Boric Acid Evaporator and Volume Control Tanks. The Waste Gas System was then filled with N_2 and H_e gases and it was leak checked using a H_e detector. The system was also sectionalized using existing system isolation valves, and pressure drop tests were taken for each section tested.
The Test proved certain theories to be erroneous and provided data useful in resolving some waste gas system problems.

Safety Evaluation:

The test proceeded uneventfully and at no time during it's course was there any circumstances that could have created an unreviewed safety question nor lowered factors of Safety of any Tech. Specs.

SPECIAL TEST 86-32

UNIT 3 VCT PURGE SPECIAL TEST

Background Information:

This special test was written and performed to provide a method to reduce the amount or waste gas system piping involved in the Volume Control Tank (VCT) purge. The waste gas compressor discharge to cover gas cross tie was isolated to preclude VCT purge gases from entering the cover gas header. The gas flow path for this special test was the same as per plant procedures, from the VCT to the waste gas compressor and to the Gas Decay Tanks. A secondary purpose of this special test was to locate the general area of waste gas system leakage if any.

Test Results:

This special test was performed and some waste gas system leakage was located in the area of the waste gas compressors.

Safety Evaluation:

This special test was found not to envolve an unreviewed safety question, because the flow path used is the same as described in the FSAR. The cross ties from the waste gas compressor discharge to the cover gas header were isolated to minimize the amount of waste gas system piping exposed to VCT purge gases and had no detrimental effect on waste gas compressor operation.

SPECIAL TEST 86-35

UNIT 4 VCT PURGE SPECIAL TEST

Background Information:

This special test was written and performed to provide a method to reduce the amount or waste gas system piping involved in the Volume Control Tank (VCT) purge. The waste gas compressor discharge to cover gas cross tie was isolated to preclude VCT purge gases from entering the cover gas header. The gas flow path for this special test was the same as per plant procedures, from the VCT to the waste gas compressor and to the Gas Decay Tanks. A secondary purpose of this special test was to locate the general area of waste gas system leakage if any.

Test Results:

This special test was performed with no significant leachages detected from the waste gas system piping.

Safety Evaluation:

This special test was found not to envolve an unreviewed safety question, because the flow path used is the same as described in the FSAR. The cross ties from the waste gas compressor discharge to the cover gas header were isolated to minimize the amount of waste gas system piping exposed to VCT purge gases and had no detrimental effect on waste gas compressor operation.

SPECIAL TEST 87-01

STEAM GENERATOR BLOWDOWN CONTROL VALVES DELAYED ACTUATION VERIFICATION TEST

Background Information:

The purpose of Special Test 87-01 is to verify the setpoints on the Agastat timer to Steam Generator Blowdown Control Valves CV-6275A, CV-6275B and CV-6275C. Each Agastat timer provides a delayed opening signal to its respective control valve following a placement of control switch (HS-6275A-1, HS-6275B-1, or HS-6275C-1) from the closed to the open position. A stopwatch will be used by the Control Room personnel to measure the time interval from the point a control switch is turned to the open position until the first indication of stem travel of the tested control valve is observed (i.e., when dual position lights of the tested valve have illuminated on the control board). Any discrepancies between the value of the setpoints indicated on the Agastat timers and the actual measured setpoints (i.e., the time intervals determined during the test) will be documented for incorporation into the system's maintenance package.

The test will also utilize field personnel stationed near the control valves. Close radio communications will be maintained between the Control Room and field personnel during test evolutions. The field personnel will verify that the pressure in the header being tested attains the secondary side steam generator pressure (as indicated on a temporarily installed pressure gauge) prior to the actuation of the tested control valve.

Status:

The procedure to perform the special test is not yet completed, but is tentatively scheduled to be available by October 1, 1987.

Test Results:

No testing has yet been performed.

Safety Evaluation:

The safety evaluation for Special Test 87-01 will be available following a completion of the written test procedure.

SPECIAL TEST 87-03

WTP, DWST AND CONDENSATE DISSOLVED OXYGEN MONITORING PROGRAM

Background Information:

Special Test 87-03 monitors dissolved oxygen levels in the make-up water system to Unit 4. High levels of dissolved oxygen in make-up water can cause corrosion in secondary system piping. Juno Beach Nuclear Services, Westinghouse and INPO recommend less than 100 ppb of oxygen in the Demineralizer Water Storage Tank (DWST). The Turkey Point Nuclear Units presently average about 2000 ppb of oxygen in their tanks.

This special test samples deoxygenated water at various points from the Water Treatment Plant (WTP) effluent to the discharge of the Condensate Pumps to Unit 4. The points sampled include the DWST, the Demineralized Water Degasifier, the Condenser Hotwells, the suction side of the Condensate Pumps, as well as, the WTP effluent and the Condensate Pump discharge. Dissolved oxygen values are obtained during normal operation of the WTP to the DWST with the Demineralizer Water Degasifier in service and with it out of service.

Status:

The test is presently awaiting the site arrival and subsequent installation of the dissolved oxygen analyzers. Installation of all equipment is scheduled for completion by August 1, 1987. The test is tentatively scheduled to be completed by the end of September, 1987.

Test Results:

Test data is scheduled to be taken in August, 1987 following the installation of all test equipment.

Safety Evaluation:

The components and sytems which make up Special Test 87-03 do not interact with any nuclear safety-related systems. The WTP is a Quality Group C system which is defined in the FSAR as "important to operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amount of radioactivity". The Condenser and Condensate Systems are Quality Group D systems while the DWST is described in the FSAR as an alternate, non-safety source of water for the Auxiliary Feedwater (AFW) pumps or the Condensate Storage Tanks (CSTs). The installation and operation of the test equipment does not affect the normal operation of these systems. Any postulated failure of the test equipment or of these systems would include tank rupture, pipe breaks, sample line breaks and valve failure. These types of accidents would not impact any equipment or system important to safety.

Based on the above, it can be said that the modifications performed under Special Test 87-03 are acceptable from the standpoint of nuclear safety. The modifications do not constitute an unreviewed safety question nor do they require a change to the Technical Specifications.

SPECIAL TEST 87-08

CONDENSER TUBING CATHODIC PROTECTION MONITORING SYSTEM

Background Information:

The purpose of the test is to obtain data on cathodic protection levels in the Unit 3 condenser waterboxes and tube sheets. This data will be evaluated by General Engineering and used to obtain safe operating levels of the cathodic protection system without causing excessive hydrogen evolution. Excessive hydrogen can be absorbed into the condenser titanium tubes causing their embrittlement and eventual failure.

The cathodic protection monitoring system installed under the special test consist of six reference electrodes mounted on each inlet and outlet tube sheet to the condenser, connecting wires that will be routed through an existing penetration in each waterbox, a junction box and a personal computer for data aquisition.

Status:

With the exception of the personal computer, all equipment associated with the monitoring system under Special Test 87-08 has been installed. The computer will be installed prior to a startup of the unit. The computer will be used to obtain the necessary data on cathodic protection levels during normal power operation.

Test Results:

Data has not been taken since the system is not yet functional and the unit is not on line.

Safety Evaluations:

The condenser, condenser waterboxes and the monitoring system installed under Special Test 87-08 perform no safety related functions. The condenser and its associated components are classified as Non-Nuclear Safety Related, Quality Group D and Non-Class 1E. The installation, operation and any postulated failures of the installed monitoring system will not interfere nor interact with any safety related equipment.

Based on the above, it can be stated that the modifications performed under Special Test 87-08 are acceptable from the standpoint of nuclear safety. The modifications do not constitute an unreviewed safety question, nor do they require a change to the Technical Specifications.

SPECIAL TEST 87-11

SAFETY INJECTION SYSTEM MOV DIFF. PRESS. STROKE TESTING
VALVES: MOV-3-843 A/B AND MOV 878 A/B

Background Information:

Test completed, awaiting complete documentation package prior to transmittal to Document Control.
NRC I & E Bulletin 85-03 item C (CTRAC # 86-0692-03) requires demonstration of MOV operability in HHSI systems required to be tested for operational readiness in accordance with 10 CFR 50.55A.

The test cycled selected valves under the maximum practical achievable differential pressure, with an SI pump operating at minimum flow conditions and suction from either the RWST or RHR pumps, as required by the design conditions established for each valve.

Test Results:

The selected valves cycled properly, without difficulty. The differential pressures achieved for each valve are as follows:

MOV-3-878, DP = 1505 psid
MOV-3-843, DP = 1585 psid

Safety Evaluation:

This test aligned the system in modes described in the FSAR and cycled valves to demonstrate operability under maximum achievable differential pressures. Therefore the probability of occurance of an accident, or the seriousness of the consequences of an accident, or the probability of malfunction of equipment, or the seriousness of the consequences of malfunction of equipment is not increased.
There were no hardware modifications required as part of this test. Therefore, there is no possibility for an accident of a different type, as possibility for a malfunction of a different type than any evaluated previously.
The test was performed within Tech. Spec. limitations, therefore the margin of safety was preserved.

SPECIAL TEST 87-13

CONDENSER TUBING STRAIN MEASUREMENT TEST

Background Information:

The purpose of Special Test 87-13 is to measure the strain of Unit 3 condenser titanium tubes following the unit's return to power after the 1987 refueling outage. Data on actual tube strain levels during full power operation is required in order to develop a more accurate criteria for future tube plugging due to hydriding (hydrogen embrittlement) of the tubes.

The special test consist of strain gauge rosettes installed on the shell side of six (6) condenser titanium tubes. Connecting cables to the gauges are mounted along the tube sheet to the 3BS outlet waterbox and encased in ARCOR 100% solids epoxy. The cables are then mounted within 1½" carbon steel, Schedule 40 piping installed along the west side and south side condenser walls, through an existing penetration and to a computer for data acquisition.

Status:

Installation of the strain gauge test equipment has been completed. However, only 5 of 6 titanium tubes were installed with strain gauges due to a lack of unplugged tubes in one desired area. The installation and operability of the strain gauge equipment was verified by the Technical Department on June 25, 1987. Tube strain data will be taken during the startup of the unit.

Test Results:

Test data has not yet been obtained but will become available following a startup of the unit.

Safety Evaluation:

The Unit 3 condenser is a Non-Nuclear Safety Related component and, as such, is not required to function during any existing analyzed accident scenarior. Therefore, modifications to the condenser as performed under Special Test 87-13 affects only Non-Nuclear Safety Related, Quality Group D equipment.

The chemical content of all materials or solutions used in preparing the test will not adversely affect any components or the secondary chemistry.

The condenser wall penetration utilized during the test and capped following the completion of the test will meet all design criteria of existing penetrations to insure that the condenser pressure boundary is maintained.

SPECIAL TEST 87-13

CONDENSER TUBING STRAIN MEASUREMENT TEST

Any postulated failures of the materials or equipment used in the test will have no impact on the safe shutdown on the plant, or on any safety related systems. Any materials associated with the test located inside the condenser which could be postulated to become dislodged would be caught in the condenser hotwell pump screen. None of these materials would be large enough to impact pump suction.

Additionally, postulated failures of the condenser would have no impact on the safe shutdown of the plant, or on any safety related systems. The condenser is not used to prevent postulated accidents, mitigate the consequences of such accidents, maintain safe shutdown conditions, or adequately store spent fuel.

Based on the above, it can be said that the modifications performed under Special Test 87-13 are acceptable from the standpoint of nuclear safety. The modifications do not consititute an unreviewed safety question nor do they require a change to the Technical Specifications.

APPENDIX A

ANNUAL REPORT OF SAFETY AND RELIEF VALVE CHALLENGES

By letter dated June 13, 1980 (L-80-186), Florida Power and Light stated the intent to comply with the requirements of Item IIK.3.3 of Enclosure 3 to the commission's letter of May 7, 1980 (Five Additional TMI-2 Related Requirements for Operating Reactors).

The following is a list of safety valve and power operated relief valve (PORV) actuations for Turkey Point Units 3 and 4 from July 1, 1986, to June 30, 1987.

Unit 3

July 15, 1986	PORV 455C and 456 were cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
July 15, 1986	PORV 455C and 456 were cycled open as per OP 0209.1, Valve Exercising Procedure. PORV operation was satisfactory.
July 24, 1986	PORV 455C and 456 were opened to provide a 2.2 square inch vent path while the reactor coolant system was depressurized as per Technical Specification 3.15. Both PORV's opened satisfactorily.
July 29, 1986	PORV 455C and 456 were cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
August 1, 1986	PORV 456 was cycled manually to test annunciator A 4/1, PORV/Relief Valve open. PORV operation was satisfactory.
December 27, 1986	PORV 456 cycled open due to high reactor coolant system pressure during a turbine runback. The valve opened but would not close. The associated block valve MOV-3-535 was closed. PORV 456 was cycled during Maintenance work. PORV operation was not satisfactory due to dual position indication.
December 28, 1986	PORV 455C was cycled twice for maintenance. PORV operation was satisfactory.
December 28, 1986	PORV 455C was cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.

December 29, 1986	PORV 455C was cycled as per Appendix B and P of OP 0209.1, Valve Exercising Procedure. PORV operation was satisfactory.
December 29, 1986	PORV 455C was cycled twice while collasping the steam bubble in the pressurizer. PORV operation was satisfactory.
January 1, 1987	PORV 455C was opened to provide a 2.2 square inch vent path while the reactor coolant system was depressurized as per Technical Specification 3.15. PORV operation was satisfactory.
January 1, 1987	PORV 455C and 456 were cycled as per 3-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. Both PORV's operated satisfactorily.
January 1, 1987	PORV 455C was cycled as per 3-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test after maintenance was performed to repair leaks found during previous test. PORV operation was satisfactory.
January 1, 1987	PORV 455C and 456 were cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
January 1, 1987	PORV 456 was cycled open as per OP 0290.1, Valve Exercising Procedure. PORV operation was satisfactory.
January 2, 1987	PORV 456 was cycled as per Temporary Procedure (TP) 292, PCV-3-456 Closure Test. PORV operation was satisfactory. This test was done as part of past maintenance testing done for PORV 456 after it did not close on December 27, 1986.
January 4, 1987	PORV was cycled open while Unit 3 was in hot standby as final operability check after maintenance completed work on the valve when it did not close on December 27, 1986. PORV operation was satisfactory.
March 7, 1987	PORV 455C was cycled open due to a misaligned adjustment pot which indicated a full pressurizer spray demand. PORV operation was satisfactory.
March 8, 1987	PORV 455C and 456 were cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.

March 9, 1987	PORV 456 cycled open during the performance of 3-OSP-050.8, RHR MOV's 750, 751, 862 and 863 Interlock Test. The test simulated a high reactor coolant system pressure and provided no precautions to prevent PORV operation when aligned in the low pressure mode of the Overpressure Mitigating System. Procedure change was completed to correct problem. PORV operation was satisfactory.
March 16, 1987	PORV 455C and 456 were cycled as per OP 0209.1, Valve Exercising procedure. PORV operation was satisfactory.
June 22, 1987	PORV 455C and 456 were cycled as per 3-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV operation was satisfactory.
June 22, 1987	PORV 455C and 456 were cycled as per 3-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
June 23, 1987	PORV 456 cycled open and did not close when power supply breaker tripped. Maintenance reset power supply and PORV 456 closed. PORV operation was satisfactory.
June 24, 1987	PORV 455C and 456 were cycled open as per 3-OP-041.4, Overpressure Mitigating System, upon completion of repairs to TE-423A. PORV operation was satisfactory.

Unit 4

July 5, 1986	PORV 455C and 456 were cycled as per 4-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
July 7, 1986	PORV 455C and 456 were cycled open as per 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test.
July 20, 1986	PORV 455C was cycled open to reduce reactor coolant system pressure. PORV operation was satisfactory.
July 27, 1986	PORV 455C and 456 were cycled open to provide a 2.2 square inch vent path while the reactor coolant system was depressurized as per Technical Specification 3.15. PORV operation was satisfactory.
August 1, 1986	PORV 455C and 456 were cycled open as per 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV operation was satisfactory.
August 1, 1986	PORV 456 cycled open while hanging in-plant equipment clearance 4-86-8-002 for safeguards rack QR80. PORV operation was satisfactory.
August 7, 1986	PORV 455C cycled open due to a high pressure condition in the reactor coolant system. PORV operation was satisfactory.
August 12, 1986	PORV 455C and 456 cycled open as per OP 1300.2, Operational Test of MOV-*-535, 536 and PORV-*-455C, 456. PORV operation was satisfactory.
August 14, 1986	PORV 455C and 456 were cycled open as per 4-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
August 15, 1986	PORV 455C and 456 were cycled open as per 4-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.

October 25, 1986	PORV 455C and 456 were cycled open as per 4-OP-041.4, Overpressure Mitigating System. PORV 456 operated satisfactory. PORV 455C was declared out of service due to dual position indication. PORV 455C was repaired and retested as per 4-OP-041.4. PORV 455C operated satisfactory.
March 11, 1987	PORV 455C and 456 were cycled open as per OP 0209.1, Valve Exercising Procedure. PORV operation was satisfactory.
March 13, 1987	PORV 455C and 456 were cycled open as per 4-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
May 12, 1987	PORV 456 was cycled as per 4-OP-041.4, Overpressure Mitigating System and 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV operation was satisfactory.
May 13, 1987	PORV 455C was cycled as per 4-OP-041.4, Overpressure Mitigating System and 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV operation was satisfactory.
May 22, 1987	PORV 455C cycled open due to an overpressure condition that resulted from PVC-4-145 not responding quickly to a reactor coolant system pressure spike. PORV operation was satisfactory.
June 10, 1987	PORV 455C and 456 were cycled as per 4-OP-041.4, Overpressure Mitigating System. PORV operation was satisfactory.
June 17, 1987	PORV 455C and 456 were cycled as per 4-OP-041.4, Overpressure Mitigating System. PORV 455C operated satisfactory. PORV 456 did not obtain proper indication. Maintenance was performed on the valve and it was retested satisfactory.

APPENDIX B

ABSTRACT

An inservice Multi-Frequency Eddy Current Examination of Steam Generators #3A, #3B and #3C at the Turkey Point Nuclear Plant was performed during the period of June 8 through June 13, 1987. The examination was conducted by the Juno Nuclear Staff - Material Codes & Inspection group and supplemented by Florida Power & Light Company certified Eddy Current personnel and complimented by contractor personnel. The examination program consisted of multi-frequency testing for detection of tube wall anomalies.

The data from the examiantion was recorded on magnetic tapes and also on computer diskettes. The recordings were evaluated by data analysts and the results recorded on computer print outs. An independent review of the data was also performed during the examination program.

The examinations were full length inspection of tubes, from the inlet side of each of the three Steam Generators.

A total of 1029 tubes (324 in #3A, 332 in #3B and 373 in #3C) were inspected. A total of three tubes were plugged. Two were non-full length tubes and were plugged electively and the third had an imperfection depth in excess of the 40% Technical Specification limit.

FORM NIS-BB OWNERS' DATA REPORT FOR EDDY CURRENT EXAMINATION RESULTS
As required by the provisions of the ASME Code Rules

SUMMARY OF EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

STEAM GENERATOR NUMBER	TOTAL TUBES INSPECTED	TOTAL INDICATIONS > OR = TO 20% TO 39%	TOTAL INDICATIONS > OR = TO 40% TO 100%	TOTAL TUBES PLUGGED AS PREVENTIVE MAINT	TOTAL TUBES PLUGGED
3E210A	324	(16) 2	0	0	0
3E210B	332	(8) 4	0	0	0
3E210C	373	(15) 3	1	2*	3

* = SHORT STUB TUBE/SHOP PLUG
(X) = < 20 %

LOCATION OF INDICATIONS

STEAM GENERATOR	AVB BARS	DRILLED SUPPORT 1 THROUGH 6		TOP OF TUBE SHEET TO 1 DRILLED SUPPORT	
		HOT LEG	COLD LEG	HOT LEG	COLD LEG
3E210A	0	(7) 0	(6) 2	(1) 0	(2) 0
3E210B	(1)	(3) 2	(2) 1	(2) 1	(0) 0
3E210C	(3)	(5) 1	(5) 1	(2) 1	(0) 1

CERTIFICATION OF RECORD

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

FLORIDA POWER and LIGHT COMPANY
(Organization)

Date: 8/28/87

By: J. T. Carr
NDE SUPERVISOR

STEAM GENERATOR TUBES PLUGGED

STEAM GENERATOR 3E210A			STEAM GENERATOR 3E210B			STEAM GENERATOR 3E210C		
ROW	COLUMN	REMARKS	ROW	COLUMN	REMARKS	ROW	COLUMN	REMARKS
0	0		0	0		7	5	SHORT TUBE STUB
						7	13	SHORT TUBE STUB
						14	89	

EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3 STEAM GENERATOR: 3E210A

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
28	20	12	HOT LEG	ABOVE FIRST SUPPORT
37	21	11	HOT LEG	ABOVE SIXTH SUPPORT
37	22	13	HOT LEG	ABOVE SIXTH SUPPORT
17	32	19	COLD LEG	ABOVE FIRST SUPPORT
37	35	11	HOT LEG	ABOVE SIXTH SUPPORT
23	48	4	COLD LEG	SLUDGE PILE
15	55	15	COLD LEG	SLUDGE PILE
9	59	13 11	HOT LEG	SLUDGE PILE ABOVE THIRD SUPPORT
29	59	15 8	COLD LEG	ABOVE FIFTH SUPPORT
9	60	19	HOT LEG	ABOVE SIXTH SUPPORT
22	60	13	COLD LEG	ABOVE THIRD SUPPORT
27	63	7	COLD LEG	ABOVE FIRST SUPPORT
40	63	2	COLD LEG	ABOVE THIRD SUPPORT

HOT LEG (INLET)
COLD LEG (OUTLET)

STEAM GENERATOR EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3 STEAM GENERATOR: 3E210A

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
4	74	19	HOT LEG	ABOVE THIRD SUPPORT
29	59	37	COLD LEG	ABOVE FIFTH SUPPORT
33	77	27	COLD LEG	ABOVE SIXTH SUPPORT

HOT LEG (INLET)
COLD LEG (OUTLET)

STEAM GENERATOR EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3 STEAM GENERATOR: 3E210B

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
5	8	12	HOT LEG	ABOVE BAFFLE
28	14	11	HOT LEG	ABOVE FIRST SUPPORT
39	25	6	HOT LEG	U-BEND
40	25	3 10	COLD LEG	ABOVE FIFTH SUPPORT
13	29	11	HOT LEG	ABOVE FOURTH SUPPORT
42	38	13	HOT LEG	SLUDGE PILE
45	45	6	HOT LEG	ABOVE FIFTH SUPPORT
5	8	22	HOT LEG	ABOVE FOURTH SUPPORT
42	37	27	HOT LEG	SLUDGE PILE
44	39	36	COLD LEG	ABOVE FIFTH SUPPORT
43	41	29	HOT LEG	ABOVE FIFTH SUPPORT

HOT LEG (INLET)
COLD LEG (OUTLET)

FORM NIS-BB OWNERS' DATA REPORT FOR EDDY CURRENT EXAMINATION RESULTS
As required by the provisions of the ASME Code Rules

STEAM GENERATOR EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3 STEAM GENERATOR: 3E210C

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
2	2	11	HOT LEG	ABOVE FIRST SUPPORT
37	21	7	HOT LEG	U-BEND
40	25	18	HOT LEG	U-BEND
27	30	15	HOT LEG	U-BEND
31	32	3	COLD LEG	ABOVE FOURTH SUPPORT
2	35	7 2	HOT LEG	SLUDGE PILE ABOVE SECOND SUPPORT
22	36	19	HOT LEG	ABOVE FOURTH SUPPORT
45	44	17	HOT LEG	ABOVE FOURTH SUPPORT
28	58	11	HOT LEG	ABOVE FIRST SUPPORT
28	58	4 8	COLD LEG	ABOVE FIRST SUPPORT
40	64	13	HOT LEG	ABOVE SIXTH SUPPORT
22	67	15	COLD LEG	ABOVE FOURTH SUPPORT
20	77	19	COLD LEG	ABOVE FIFTH SUPPORT

STEAM GENERATOR EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT NUCLEAR POWER PLANT UNIT NO. 3 STEAM GENERATOR: 3E210C

EXAMINATION DATES: JUNE 8, 1987 THRU JUNE 13, 1987

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
5	1	38	COLD LEG	ABOVE FIFTH SUPPORT
7	13	36	HOT LEG	SLUDGE PILE
3	88	28	HOT LEG	ABOVE BAFFLE
14	89	48	COLD LEG	SLUDGE PILE

HOT LEG (INLET)
COLD LEG (OUTLET)

P. O. BOX 14000, JUNO BEACH, FL 33408-0420

FPL

AUGUST 3 1 1987

L-87-365

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

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
10 CFR 50.59 Report

Florida Power & Light's Report on "Changes, Tests and Experiments Made Without Prior Commission Approval" for the period July 1, 1986 through June 30, 1987 is attached.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/SDF/gp

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

IE47
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SDF1/061/1