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, 1985 THROUGH JUNE 30, 1986

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

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, 1985 through June 30, 1986.

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, 1985 to June 30, 1986.

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, 1985 through June 30, 1986.

TITLE 10, SECTION 50.59 CFR REPORT

(i) COMPLETED PC/M LIST

JULY 1, 1985 THROUGH JUNE 30, 1986

PC/M NO.	PC/M TITLE DESCRIPTION	UNITS COMPLETED THIS PERIOD
78-101B	Steam Generator Blowdown Recovery System	3
80-42	Addition of Steam Generator Access Platform	3
81-76	Seismograph Relocation	3 and 4
81-111	Replace Fisher Control Co. Valve Positioner	3
81-146	Pressurizer Spray Valves and Pipe Relocation	3
82-76	Add Bay Position #10 for Florida City Line	Common
82-320	Add Containment Purge Valve Debris Screens	3
83-96	Pressurizer Heater Ammeters	3
83-169	PASS Flow Indication	3 and 4
83-201	Loose Parts Metal Impact Monitor System	3
84-208	Reinstate of Power Mismatch	3
84-219	SFP Rack Indexing System	3
85-049	Repair ICW Pipe Inst., Vent, Drain Conn.	3
83-185	Computer Repair and Supv./Program Facility	3 and 4
84-19	Demin. Water Supply Line to Lab Sink	3 and 4
84-87	Transfer Air Particulate & Gas Monitor Fan 3V36	3
84-89	Containment Purge Valve Bolts	3
84-116	Modification to CVCS (I.C.) Per I.E.B. 79-14 (11B)	3
84-117	Modification to CVCS (I.C.) Per I.E.B. 79-14 (11C)	3
84-118	Modification to CCW (I.C.) Per I.E.B. 79-14 (29)	3
84-160	Modification to CCW (I.C.) Per I.E.B. 79-14 (17)	3
84-164	Modification to S.I. & RHR (I.C.) Per I.E.B. 79-14 (10)	3
80-28	Inst. Run Time Meters on Charging Pump	4
85-38	Inspect and Repair ICW Piping	3
85-75	RCP 3C Electrical Penetration Canister Repair with Upgrade	3
82-98	Aux. Power Upgrade, Addition of Transformers, Switchgear and Load Centers	4
84-01	Modification to Comply with Reg. Guide 1.97, Rev. 3 Qualified Limit Switches	3

TITLE 10, SECTION 50.59 CFR REPORT

COMPLETED PC/M LIST

JULY 1, 1985 THROUGH JUNE 30, 1986

PC/M NO.	PC/M TITLE DESCRIPTION	UNITS COMPLETED THIS PERIOD
83-76	Fire Water Pipe Hangers	3
84-210	Turbine Runback Modifications	3
85-80	Cathodic Protection Control Cabinet Rheostat Removal	4
85-100	ICCS - QSPDS Software Modification	3
85-092	Replace Teledyne - Farris CCW Relief Valves	3
85-112	Containment Purge Supply Valves Mech. Stop	3
85-104	Comp. Substitution for Valve 3-20-736	3
85-48	Repair ICW Basket Strainer	3
84-202	MCC Changes and Constant Voltage	3 and 4
84-205	Raceways for Inverter Replacement	3 and 4
80-119	Charging Pump System Improvement	3
85-03	Reactor Trip Breaker Auto Shunt Trip	3
85-59	Env. Sealing of Pyco RTDS (Cont. Filt)	3
83-128	Replace Class 1E AC Motor Act for NS Valves	3
81-137	Control Rod Drive Mechanism Cooler Hx Replacement	3
82-179	SFP Cooling Pump Suction Vortex Diffuser	3
84-217	Modification to S.I. and RHR (O.C.) Per I.E. 79-14	3
85-57	Steam Generator Blowdown System FCV/Bypass Valve Interlock	3
85-40	Env. Sealing of Pyco RTDS	3
84-24	Replace Flow Switches on Charcoal Filter Bed Fans	3
82-126	Modification to RCS (I.C.) Per IEB 79-14	3
83-138	Water Supp. for Appendix R Modification	3 and 4
83-178	Negative Sequence Relay Timer	3
83-190	Main Steam Line FT 485 Tubing Support Modification	3
84-08	MSIV/CV4-204 Power Supply	4
84-59	Add Steel Guards for H_2 Monitor AE-3-6307	3
84-60	Add Steel Guards for H_2 Monitor AE-4-6307	4
84-106	Modification to CVCS (I.C.) Per IEB 79-14	3

TITLE 10, SECTION 50.59 CFR REPORT

COMPLETED PC/M LIST

JULY 1, 1985 THROUGH JUNE 30, 1986

PC/M NO.	PC/M TITLE DESCRIPTION	UNITS COMPLETED THIS PERIOD
84-220	Source and Intermediate Range Detector Weep Holes	3
85-21	Env. Sealing Pyco RTD on RCS Hot/Cold Leg Loops	4
80-42	Add Steam Generator Access Platform	3
82-15	Modification to RCS (I.C.) Per IEB 79-14	3
82-95	Fisher and Porter Transmitter Replacement	3
84-51	Reactor Cavity Sump Access Door Replacement	3
84-57	Install Swtichyard Breaker 6B in Bay 6	3
84-84	Transfer CRDM Cooler Fans 3A-3B	3
84-92	Air Dryer Steam Jet Air Ejector Radiation Monitor	4
84-132	Vital Power Supply PC 600, 601, Control Relays	3
84-133	Vital Power Supply PC 600, 601, Control Relays	4
84-202	MCC Changes and Constant Voltage Transformer for Inverter Replacement	3 and 4
85-115	Reactor Trip Breaker Auto Shunt Trip	3
84-127	Addition of D/G Cooling System Instr.	3 and 4
82-296	Standby Steam Generator Feed Pump	3 and 4
83-65	RHR and HHSI Valve Position Indication	3
83-69	Reactor Head Shielding	3
83-115	Pressurizer Heater Ammeter	4
83-136	Water Suppression System for Appendix R	3
83-137	Water Suppression System for Appendix R	4
84-129	Gland Flange Repairs for 3/4" Rockwell Edwards Valve (T58)	3 and 4
84-168	Lockout of D/G Power to the Motor Driven Fire Pump	3 and 4
82-296	Standby Steam Generator Feed Pump	3 and 4
84-63	Replace D/G Control Room Isolation Switch	3
84-64	Replace D/G Control Room Isolation Switch	4
84-157	Replace Type CFD Differential Relay in "A" EDG	3
81-110	Health Physics Control Facility	3 and 4

TITLE 10, SECTION 50.59 CFR REPORT

COMPLETED PC/M LIST

JULY 1, 1985 THROUGH JUNE 30, 1986

PC/M NO.	PC/M TITLE DESCRIPTION	UNITS COMPLETED THIS PERIOD
84-148	Boric Acid Transfer Pump Drain Valve	4
85-08	Modification to Covered Walkway from Aux. Building to Containment Entrance	4
85-58	Steam Generator Blowdown System FCV/Bypass Valves Interlock	4
85-113	Steam Generaor Blowdown Isolation Valve Time Delay Relay Modification	4
82-123	Accumulator Level Transmitter Modification	3
84-151	TPCW/CCW Cathodic Protection	3
84-152	TPCW/CCW Cathodic Protection	4
82-163	Existing Fire Tank System Modification	4
84-27	Dedicated Fire Protection System Modification	3 and 4
84-147	Boric Acid Transfer Pump Drain Valve	3
85-103	NIS Input to Turbine Runback	4
82-236	Containment Purge Valve Seats and Hub Seal Replacement	3
83-64	Improved Floor Drains for Containment Spray Pump Room	4
CPWO 84-143	Replace GE 12HFA51A42F Relays	4
85-46	SFP Tool Storage Rack Modification	3
86-08	SFP Camera Monitors	4
CPWO 86-13	AFW Mechanical Overspeed Trip Device	3 and 4
84-205	Raceway for Inverter Replacement	3 and 4
85-79	Cathodic Protection Control Cabinet Rheostat Removal	3
85-111	Steam Generator Blowdown FCV Timing Modification	3
80-83	Boric Acid Transfer Pump Replacement	3 and 4
83-84	Add RWST Level Indication	4
CPWO 84-61	Refueling Crane Limit Switch	4
84-139	Boric Acid Batching Tank Agitator Replacement	3 and 4
84-197	Backup Bottled N_2 Supply to MSIV's	4
84-192	Modification to CCW (O.C.) Per IEB 79-14 CCW47	3
85-164	ICW Pump Check Valve Air Closing Cylinder Clevis Pins	3

PLANT CHANGE/MODIFICATION 78-101B

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/25/85

SUMMARY DATE: 07/29/85

REVISION: 1

Steam Generator Blowdown Recovery System

Summary:

The Steam Generator Blowdown Recovery System was primarily designed to maintain the required chemistry of the steam generator by providing means for the removal of the foreign matter which normally concentrates in the evaporator section of the S.G. The S.G.B.R.S. was part of the Steam Generator Protection Plan. The S.G. Protection Plan had been devised to preclude S.G. problems from original design.

Safety Evaluation:

The modification of Steam Generator Blowdown Recovery System does not represent an unreviewed safety question. This modification would not decrease any margin of safety discussed in the Technical Specifications. In actuality, the margin of safety will be increased due to the upgraded piping design. No changes in the Technical Specifications is warranted as the dose limit to the liquid radwaste is not affected.

PLANT CHANGE/MODIFICATION 80-42

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/30/85

REVISION: 0

ADDITION OF STEAM GENERATOR ACCESS PLATFORM

Summary:

This modification provided access to the steam generator manway cover and to support a manway cover handling device and a manway cover storage compound.

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 81-76

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3 AND 4

TURNED OVER DATE: 07/17/85

SUMMARY DATE: 07/30/85

REVISION: 0

SEISMOGRAPH RELOCATION

Summary:

This PC/M consisted of the relocation and installation of the "SMA - Strong Motion Accelerograph". The SMA-1 was removed from Unit No. 3 North Tendon Inspection Pit and installed in the Unit No. 3 South Electrical Penetration Room. The relocation was implemented due to the poor environmental conditions which exist in the Tendon Inspection Pit. Since the SMA-1 was installed below existing cable tray and metal platform, the existing penetration room lighting was insufficient for normal periodic maintenance. Therefore, localized lighting and power source for the SMA-1 battery charger was provided.

Safety Evaluation:

The "SMA-1 Supports and Components" do not perform a safety function. However, since they have the potential of interacting with safety related components in their vicinity, their supports have been designed to withstand the maximum earthquake loading (E') used for the design of Turkey Point 3 and 4 Seismic Category I Structures, in accordance with Appendix 5A of the FSAR. This design has been treated as a Safety Related Design Feature in accordance with EPP-QI 23. 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 previuosly evaluated. Therefore, it can be concluded that this PC/M does not pose any unreviewed safety question.

PLANT CHANGE/MODIFICATION 81-111 PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/30/85

REVISION: 0

REPLACE FISHER CONTROL COMPANY VALVE POSITIONER

Summary:

This is a generic CPWO which allows use of the Fisher Model 3582 valve positioner in place of the disconnected Model 3560.

Safety Evaluation:

This change is not nuclear safety related nor does it involve an unreviewed safety question. The new positioner is not used in any safety related system. It is used in the feed and condensate systems for positioning of control valves.

PLANT CHANGE/MODIFICATION 81-146 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/20/85

REVISION: 0

PRESSURIZER SPRAY VALVES AND PIPE RELOCATION

Summary:

This modification relocated the pressurizer spray control valves PCV4-455A and 4-455B from their harsh environment at the 73 foot level down to the 16 foot 3 inch and 22 foot elevations respectively. They were also equipped with isolation valves 4-572 and 4-573 to facilitate inspection and maintenance of 455A and 455B.

Safety Evaluation:

This nuclear safety related modification did not change the essential make-up of the spray piping but merely relocated the spray valves for maintainability purposes. The addition of two passive maintenance valves (locked open) did not appreciably change the pressure - boundary involved in the original analysis. Additionally, since no active components have been added by the modification, the probability or consequences of malfunction of equipment important to safety remains as previously evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 82-76

PC/M CLASSIFICATION: NNS

UNIT: COMMON

TURNED OVER DATE: 07/11/85

SUMMARY DATE: 07/30/85

REVISION: 0

ADD BAY POSITION NO. 10 FOR FLORIDA CITY LINE

Summary:

This PC/M installed Bay Position No. 10 and associated relay equipment to the PTP switchyard.

Safety Evaluation:

This modification does not perform a safety related function nor is it used as an input for a safety related function. It does not change or affect any previous safety analysis. This PC/M is non-safety related and does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 82-320

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/30/85

REVISION: 0

ADDITION OF CONTAINMENT PURGE VALVE DEBRIS SCREENS

Summary:

New seismically designed ducting containing debris screens were installed on the containment purge supply and exhaust valves inside containment. The debris screens were added to satisfy NRC concerns regarding potential blockage of valves in the open position during postulation.

Safety Evaluation:

The debris screens, duct spool piece and associated supports are safety related and seismically designed. This indication does not involve any unreviewed safety questions. This modification will enhance the margin of safety in the system by protecting the valves from being blocked by postulated debris while closing.

PLANT CHANGE/MODIFICATION 83-96

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 06/21/85

SUMMARY DATE: 07/29/85

REVISION: 0

PRESSURIZER HEATER AMMETERS

Summary:

This PC/M added an ammeter, phase selector switch, and an additional current transformer to each of the three pressurizer heater group breakers. This will provide a way to determine if one of the heater groups is not functioning properly.

Safety Evaluation:

The pressurizer heater ammeters and CT's installed by this PC/M are not safety related nor do they serve as an input to a S.R. function. The probability of occurrence or the consequences of a design basis or malfunction of Equipment Important to the Safety of the plant has not been increased. Therefore, it can be concluded that this PC/M does not pose an unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-169

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3 AND 4

TURNED OVER DATE: 07/01/85

SUMMARY DATE: 07/29/85

REVISION: 1

POST ACCIDENT SAMPLING SYSTEM (PASS) FLOW MODIFICATION

Summary:

The subject PC/M modified PASS to increase the flow to components in the system. The tubing between the pressure regulator sampling equipment and the sample pump was increased to 1/2" O.D. from 1/4" O.D. Additionally, the pump was replaced with a larger capacity pump and associated valves, fittings, and tees were changed out to a larger size compatible with the 1/2" tubing.

Safety Evaluation:

This is a non-nuclear safety related change. The equipment installed meets or exceeds the qualification of components already in use, or replaced by this PC/M, in the system. No safety related equipment was affected. No increase in the probability of an accident results. No new possibility for an accident of a different type results. No margin of safety as defined in the basis for any Technical Specification was reduced. Hence, no unreviewed safety question results.

PLANT CHANGE/MODIFICATION 83-201

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 06/25/85

SUMMARY DATE: 07/30/85

REVISION: 0

LOOSE PARTS METAL IMPACT MONITORING SYSTEM - UNIT 3

Summary:

This PC/M installs a 13 channel metal impact monitor on the Unit 3 Reactor Coolant System and Steam Generators. A system alarm is provided to DDPS and on Annunciator panel G.

Safety Evaluation:

This change is non-nuclear safety related because this system is installed for monitoring only and is not needed to mitigate the consequences of any accident in the FSAR.

PLANT CHANGE/MODIFICATION 84-208

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/17/85

SUMMARY DATE: 07/29/85

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 84-219 PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/30/85

REVISION: 0

SPENT FUEL POOL RACK INDEXING SYSTEM

Summary:

The PC/M placed channel on the North wall and on the bridge crane trolly supports. These channels had plastic cards with identification marks attached to them. The PC/M provided a new indexing system for the SFP rack up-date. No NRC requirement/commitment is involved.

Safety Evaluation:

The indexing system is not nuclear safety related. No unreviewed safety question is involved. The indexing system is not required to safely move fuel, and the channels have been installed seismically so as to prevent impact on the safety related equipment.

PLANT CHANGE/MODIFICATION 85-049

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 07/30/85

REVISION: 0

REPAIR ICW PIPE INSTRUMENTS, VENT AND DRAIN CONNECTIONS

Summary:

This modification consisted in inspecting and repairing instrument, vent and drain connections on the ICW piping and equipment which was corroded.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since no probability/consequence of an accident/malfunction is increased. Also, the margin of safety as defined in the basis for any Technical Specification will not be reduced.

PLANT CHANGE/MODIFICATION 80-28

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 07/03/85

SUMMARY DATE: 08/12/85

REVISION: 0

INSTALLATION OF RUN TIME METERS ON THE CHARGING PUMPS

Summary:

This modification installed run time meters on the charging pump motors to monitor the actual running time of the pumps. This will allow to evaluate bearing performance, performance of pump seals, etc., and allowing a more accurate planning of periodic maintenance.

Safety Evaluation:

This modification is Nuclear Safety with no Unreviewed Safety Question, since the probability of occurrence/consequences of an accident/malfunction of equipment important to safety has not increased nor has the possibility for an accident/malfunction of a different type. Thus, the margin of safety as defined in the Basis for Technical Specifications has not decreased.

PLANT CHANGE/MODIFICATION 85-38

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/24/85

SUMMARY DATE: 08/12/85

REVISION: 0

INSPECTION AND REPAIR OF ICW PIPING

Summary:

This PC/M provided for details for the inspecting, cleaning, and repairing of above Ground Intake Cooling Water (ICW) piping from the ICW pumps to the Header Isolation Valves (3-308 and 3-310). This modification was a result of recent ultrasonic and visual inspections that indicated erosion/corrosion in the ICW System.

Safety Evaluation:

This modification is Nuclear Safety with no Unreviewed Safety Questions, since there is no change in the probability or consequence of an accident on equipment malfunction as previously evaluated in the FSAR. With respect to the margin of safety as defined in the Basis for any Technical Specification, no margin will be reduced.

PLANT CHANGE/MODIFICATION 85-75

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 07/11/85

SUMMARY DATE: 08/12/85

REVISION: 0

RCP 3C ELECTRICAL PENETRATION CANISTER REPAIR WITH UPGRADE

Summary:

This modification replaced cracked electrical insulator bushing located outside containment in the 5KV electrical penetration canister serving RCP 3C with an upgraded bushing.

Safety Evaluation:

The 5KV electrical penetration canisters are safety-related because they are considered an integral part of an unfired pressure vessel by the original design specification.

Accordingly,

a. The probability of occurrence or the consequences of design basis accidents or malfunction of equipment important to safety previously evaluated in the FSAR is not increased since the penetration is required to satisfactorily pass a leak rate test with acceptance criteria based on ASME Boiler and Pressure Vessel Code Section XI requirements.

b. The possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR is not created since Containment integrity has already been evaluated in Section 14.3.4 of the FSAR.

c. The margin of safety as defined in the basis for any Technical Specification is not reduced. Satisfactory completion of the leak rate test will satisfy the margin of safety defined in the basis for Technical Specification 4.4 Containment Test.

PLANT CHANGE/MODIFICATION 82-98

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 03-06-85

SUMMARY DATE: 08-28-85

REVISION: 2

AUXILIARY POWER UPGRADE, ADDITION OF TRANSFORMER, SWITCHGEAR AND LOAD CENTERS

Summary:

This PC/M installed the high voltage fly-tap connection, circuit int. SW, C-Bus transformer, 4.16KV outdoor switchgear and 480V load centers for Unit No. 4.

Safety Evaluation:

Since this PC/M is NNS and its design insured minimum interfaces with safety related equipment, it does not involve an unreviewed safety question 1) with the probability of occurrence of an accident previously evaluated in the FSAR, 2) with respect to the possibility of an accident of a different type than any analyzed in the FSAR, or 3) with respect to the possibility of malfunction of a different type than analyzed in the FSAR.

PLANT CHANGE/MODIFICATION 84-01 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/10/85

SUMMARY DATE: 09/04/85

REVISION: 1

MODIFICATIONS TO COMPLY WITH REG. GUIDE 1.97 REV. 3 REQUIRED TO PROVIDE QUALIFIED LIMIT SWITCHES

Summary:

This PC/M replaced the limit switches for the RCDT and the CCW Cont. Isol. Valves with fully qualified Namco switches.

Safety Evaluation:

This PC/M is safety related. The one for-one replacement of the existing safety related limit switches with qualified Namco switches will extend the environmental and electrical integrity of the existing limit switches. Therefore, the consequences of an accident previously analyzed in the FSAR would not be altered.

PLANT CHANGE/MODIFICATION 83-76

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 08-22-85

SUMMARY DATE: 09-09-85

REVISION: 0

FIRE WATER HANGER - PIPE HANGERS

Summary:

This design package provides for the replacement of loose and broken pipe hangers in the Fire Protection System. It also provided new pipe hangers in locations where excessive unsupported spans exist.

Safety Evaluation:

The Fire Water System has no nuclear safety function and its failure will not adversely impact nuclear safety related items. Therefore, it can be concluded that this PC/M does not pose any unreviewed safety questions.

PLANT CHANGE/MODIFICATION 84-210

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 08-29-85

SUMMARY DATE: 09-09-85

REVISION: 0

TURBINE 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 85-80

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 08-29-85

SUMMARY DATE: 09-09-85

REVISION: 0

CAT. PROT. CONTROL CABINET RHEOSTAT REMOVAL

Summary:

This PC/M replaced the rheostat in the intake, condenser and containment cat. prot. systems and installed jumpers from the bus bar to the ammeter shunts.

Safety Evaluation:

This PC/M is not safety related. This modification does not affect any safety related system in the plant. The operation of the cathodic protection system will be enhanced by the replacement of the rheostats with jumpers.

PLANT CHANGE/MODIFICATION 85-100

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 09/20/85

SUMMARY DATE: 09/30/85

REVISION: 0

ICCS - QSPDS SOFTWARE MODIFICATIONS

Summary:

This PC/M modified the QSPDS software to improve the operator interpretation of the ICCS display pages and to be able to identify an inadequate core cooling situation. No hardware changes were involved thus maintaining all information complying with NUREG-0737 Section II-F-2.

Safety Evaluation:

The QSPDS software modification is Nuclear Safety Related because the equipment is to display information to advise the operator of an Inadequate Core Cooling situation, which is a Safety Related event.

This changes does not constitute an unreviewed safety question because: The probability of occurrence, or the consequence of an accident or malfunction important to safety, previously evaluated in the FSAR, has not been increased. The Inadequate Core Cooling System does not perform any direct or indirect equipment actuation, it is only for display information. The software changes improve the operator abilities to detect and follow the accident situation. Parameters as Subcooling Margin, Core Exit Temperature, and Reactor Vessel Level can be easily monitored with this system.

For the same reasons, no possibility for an accident or malfunction of a different type from any evaluated previously in the FSAR has been created by this modification. Additionally, the margin of safety, as defined in the bases for Technical Specifications, has not been decreased.

Therfore, it is concluded that the software modification does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-185

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3 AND 4

TURNED OVER DATE: 03/18/85

SUMMARY DATE: 08/01/85

REVISION: 0

COMPUTER REPAIR AND SUPERVISOR/PROGRAMMER FACILITY

Summary:

This PC/M added a small building with two rooms onto the computer room at the 18 foot elevation. This extension of the control building between Units 3 and 4 will be for the computer programmer's office and an I and C repair shop.

Safety Evaluation:

The new computer repair and supervisor/programmer facility does not perform a safety function or provide protection for safety related systems or equipment. The building's power supply is from a non-safety related source and all conduit installed under this PC/M will be mounted seismically to prevent adverse impact on any safety related equipment that is near. As discussed in the Design Analysis, the control building will not be adversely affected by the new structure. Therefore, no unreviewed safety questions are posed and no margin of safety decreased by this PC/M.

PLANT CHANGE/MODIFICATION 84-89 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/03/85

SUMMARY DATE: 08/01/85

REVISION: 0

CONTAINMENT PURGE VALVE BOLTS

Summary:

This modification replaced the existing operator and trurnion bolts, including respective nuts, on the 54" containment purge valves POV-3-2602, POV-3-2603, POV-4-2602 and POV-4-2603. The replacement bolts were of SAE Grade 8 carbon steel and the replacement nuts were heavy hex of ASTM A194 Grade 2H carbon steel. The original bolts and nuts were SAE Grade 2 carbon steel.

Safety Evaluation:

This design package is nuclear safety related because containment purge valves function to perform/maintain containment isolation. Stress reports show that the new components are better than the originals and appropriate for substitution. Therefore, the use of the new bolts and nuts is acceptable and does not constitute an unreviewed safety question.

PLANT CHANGE/MODIFICATION 84-116

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/20/85

SUMMARY DATE: 08/01/85

REVISION: 0

MODIFICATION TO CVCS SYSTEM (I.C.) PER I.E. BULLETIN 79-14, CVCS-11B

Summary:

This PC/M modified the pipe supports in the Chemical and Volume Control System inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-117

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/08/85

SUMMARY DATE: 08/01/85

REVISION: 0

MODIFICATION TO CVCS SYSTEM (I.C.) PER I.E. BULLETIN 79-14, CVCS-11C

Summary:

This PC/M modified the pipe supports in the Chemical and Volume Control System inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-118

PC/M CLASSIFICATION:	NS
UNIT:	3
TURNED OVER DATE:	06/11/85
SUMMARY DATE:	08/01/85
REVISION:	0

MODIFICATION TO CVCS SYSTEM (I.C.) PER I.E. BULLETIN 79-14, CCW-29

Summary:

This PC/M modified the pipe supports in the Component Cooling Water inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-160

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/08/85

SUMMARY DATE: 08/01/85

REVISION: 0

MODIFICATION TO CCW SYSTEM (I.C.) PER I.E. BULLETIN 79-14, CCW-017

Summary:

This PC/M modified the pipe supports in the Component Cooling Water System inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-164

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/07/85

SUMMARY DATE: 08/01/85

REVISION: 0

MODIFICATION TO S.I. AND RHR SYST. (I.C.) PER I.E. BULLETIN 79-14, 010

Summary:

This PC/M modified the pipe supports in the Safety Injection and Residual Heat Removal System inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-19 PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3/4

TURNED OVER DATE: 02/27/85

SUMMARY DATE: 08/01/85

REVISION: 0

DEMINERALIZED WATER SUPPLY LINE TO LAB SINK

Summary:

This modification added a demineralized water supply line from a hose connection in the Hot Machine Shop (Valve 10-488) and tie into an existing supply line from the Lab Demineralized Water Tank. The purpose of this modification was to provide a more reliable means of supplying demineralized water to the Chemistry Lab since the tank cannot be filled due to a reduced supply pressure in the header. This was an Employee Suggestion Number 5606 that recommended this modification.

Safety Evaluation:

The subject modification is non-nuclear safety related. The intent is to provide a more reliable means of demineralized water to the Chemistry Hot Lab.

No safety related system, component, or structure will be affected by this change. Therefore, this non-nuclear safety related change will not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously analyzed in the FSAR.

2) Create the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR.

3) Reduce the margin of safety as defined in the basis for any Technical Specification.

Therefore, this change does not propose an unreviewed safety concern.

PLANT CHANGE/MODIFICATION 84-87

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/11/85

SUMMARY DATE: 08/01/85

REVISION: 2

TRANSFER OF AIR PARTICULATE AND GAS MONITOR FAN 3V36

Summary:

The containment and plant vent air particulate and radioactive gas monitors (R3-11 and R3-12 measure air particulate and gaseous gamma radioactivity inside containment. This modification provides for the transfer of power from a non-vital to a vital bus.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question since the probability of occurrence of an accident/malfunction of equipment preciously evaluated in the FSAR will not increase nor will the consequences for an accident increase. Therefore, the margin of safety was not decreased as mentioned in the Technical Specifications.

PLANT CHANGE/MODIFICATION 85-092

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 08-08-85

SUMMARY DATE: 09-30-85

REVISION: 0

REPLACEMENT OF TELEDYNE-FARRIS CCW RELIEF VALVES

Summary:

This modification consisted in replacing relief valves 3-1430 through 3-1435 (CCW) Model 1870 with new valves (Model 1850) since originals were damaged and parts were not available.

Safety Evaluation:

There is no change in the probability of occurrence or the consequences of an accident previously evaluated in the FSAR. Since there is no change to the function of the relief valves on the CCW sytem, the probability of occurrence of an accident not previously evaluated in the FSAR is not changed. There is no impact on Technical Specifications or margins to safety limits.

Therefore it is concluded that the replacrement of CCW relief valves 3-1430 through 3-1435 does not pose any unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-112

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 08-08-85

SUMMARY DATE: 09-30-85

REVISION: 0

CONTAINMENT PURGE SUPPLY VALVES MECHANICAL STOP UPGRADE

Summary:

This modification upgraded the existing robot arm actuator mechanical stops on the containment purge supply valve which was deformed during recent valve cycling. Upgrade included the installation of stops of greater diameter to ensure against any further deformation during the valve open mode.

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 85-104

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 09-20-85

SUMMARY DATE: 09-30-85

REVISION: 0

COMPONENT SUBSTITUTION FOR VALVE 3-20-736

Summary:

This modification consisted in replacing existing Kerotest valve with a new Whitey valve with similar characteristics. Original was damaged and valve was not available through the manufacturer.

Safety Evaluation:

It can, therefore, be concluded that operation of the steam generator level stant pipe isolation valve with the substitution does not increase the consequences or probability of occurrence of an accident or malfunction of equipment to safety previously evaluated in the FSAR. Operation in this mode is addressed in the FSAR and, therefore, does not introduce an accident or malfunction of equipment not previously evaluated. The margin of safety as defined in the basis for any Technical Specification would not be appreciably increased or decreased by this mode of operation.

PLANT CHANGE/MODIFICATION 85-48

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 08-27-85

SUMMARY DATE: 09-30-85

REVISION: 0

REPAIR OF ICW BASKET STRAINER

Summary:

This modification provided for the inspection, cleaning and repair of the intake cooling water (ICW) basket strainers located in the component cooling heat exchanger area. Purpose of the work was to perform repair on the sealing surfaces of the strainers affected by corrosion.

Safety Evaluation:

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

PLANT CHANGE/MODIFICATION 84-202

PC/M CLASSIFICATION: NS

UNIT: 3 AND 4

TURNED OVER DATE: 08-08-85

SUMMARY DATE: 09-30-85

REVISION: 0

MCC CHANGES & CONSTANT VOLTAGE TRANSFORMERS FOR INVERTER REPLACEMENT

Summary:

This PC/M provided a regulated 120V AC backup supply for each of the normally operating vital inverters.

Safety Evaluation:

This PC/M is safety related. The probability of occurrence of an accident previously evaluated in the FSAR will not be increased since it does not change the function of any plant system, and the loading of the vital busses is not increased. Also the consequence of an accident previously evaluated in the FSAR will not be affected.

PLANT CHANGE/MODIFICATION 84-205

PC/M CLASSIFICATION: NS

UNIT: 3 AND 4

TURNED OVER DATE: 08-08-85

SUMMARY DATE: 09-30-85

REVISION: 0

RACEWAYS FOR INVERTER REPLACEMENT

Summary:

This PC/M installed conduits, junction boxes and trays as well as foundation for constant voltage transformers (CVT), transfer switches, and synchronization switches for PC/M 83-117 inverter replacement.

Safety Evaluation:

This PC/M is safety related. All raceways and equipment were seismically supported. The probability of occurrence of an accident previously evaluated in the FSAR will not be increased. There is no possibility that an accident will be created which is of a diferent type than already evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 80-119

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/11/85

SUMMARY DATE: 09/23/85

REVISION: 0

CHARGING PUMP SYSTEM IMPROVEMENTS

Summary:

This PC/M provides for a charging pump recirculation line and a thorttling valve to provide for low pressure break-in of the charging pumps after maintenance.

Safety Evaluation:

This PC/M hsas no detrimental effect on any plant system or component important to safety and does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-03

PC/M CLASSIFICATION: NS-SO

UNIT: 3

TURNED OVER DATE: 08-13-85

SUMMARY DATE: 09-09-85

REVISION: 0

RX TRIP BREAKER AUTO SHUNT TRIP

Summary:

This modification provided an automatic trip feature to the reactor trip breaker shunt coil and separated the tripping and closing circuits. This was an NRC commitment per NRC Generic Letter 83-28 Item 4.3 requiring that the reliability of reactor trip system be enhanced to also use the shunt trip attachments to open the reactor trip breaker automatically.

Safety Evaluation:

This modification did not involve an unreviewed safety question because the probability of occurrence/consequences of an accident previously evaluated in the FSAR is not affected. With respect to consequences/possibility of an accident/malfunction has not changed nor was the margin of safety as defined in the Basis for any Technical Specification.

PLANT CHANGE/MODIFICATION 85-59

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/20/85

SUMMARY DATE: 10/02/85

REVISION: 0

ENVIRONMENTAL SEALING OF PYCO RTD'S (CONTAINMENT FILTERS)

Summary:

This modification involved the application of an epoxy sealing compound on the containment charcoal filter unit RTD's. This was performed to increase the watertight integrity of the subject RTD assemblies, thus increasing the qualified life of the subject RTD's to 40 years.

Safety Evaluation:

This PC/M does not involve an unreviewed safety question because it has no detrimental effect on any plant system or component important to safety and no margin of safety as defined in the Bases for any Technical Specifications was reduced.

PLANT CHANGE/MODIFICATION 83-128

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/05/85

SUMMARY DATE: 10/02/85

REVISION: 0

REPLACEMENT OF CLASS IE AC MOTOR ACT. FOR S.R. VALVES

Summary:

This PC/M replaced six Limitorque AC powered motor actuators for safety related valves located inside the containment since they were not qualified for inside containment use.

Safety Evaluation:

The new actuators are the same as those being replaced but qualified for the environmental conditions of the cont. area. Therefore, no system characteristics will be changed and the probability of an accident would be no greater. The consequences of an accident previously analyzed in Ch. 15 of the FSAR would not be altered. The consequences of equipment malfunction are no more severe than previously analyzed in Ch. 15.

PLANT CHANGE/MODIFICATION 81-137	PC/M CLASSIFICATION: NS
	UNIT: 3
	TURNED OVER DATE: 06/17/85
	SUMMARY DATE: 10/02/85
	REVISION: 0

CONTROL ROD DRIVE MECHANISM COOLER HEAT EXCHANGER REPLACEMENT

Summary:

This PC/M replaces the heat exchangers in the control rod drive mechanism (CRDM) coolers. The function of the CRDM coolers is to remove heat losses from the CRDM's. Ambient containment air is drafted across the drive mechanisms where it gathers heat. This air is then drawn through the heat exchangers, cooled by component cooling water, and returned to containment.

Safety Evaluation:

This heat exchanger replacement does not involve an unreviewed safety question because:

No accident was evaluated in the FSAR stemming from failure of the CCW system, therefore, the probability of an accident previously evaluated in the FSAR is unchanged. The pressure boundary of the CCW system is unchanged, the CRDM coolers were not used to mitigate any accident evaluated in the FSAR and the location of the CRDM coolers precludes interaction with other systems. This replacement then does not increase the consequences of any accident previously evaluated in the FSAR, therefore, the consequences of the malfunction of equipment importatnt to safety previously evaluated in the FSAR have not been increased. The possibility of malfunction of equipment important to safety of a different type than any analyzed in the FSAR is not increased. No decrease in any margin of safety.

PLANT CHANGE/MODIFICATION 82-179

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/01/85

SUMMARY DATE: 10/02/85

REVISION: 0

SFP COOLING PUMP SUCTION VORTEX DIFFUSER

Summary:

This PC/M modified the high suction to the SFP cooling pump by adding a horizontal tee on the pipe inlet to prevent the formation of a vortex at the top of the pool. This vortex has previously caused cavitation in the pumps.

Safety Evaluation:

This PC/M does not have a detrimental effect on any plant system or component important to safety and therefore does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 84-217

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/11/85

SUMMARY DATE: 10/02/85

REVISION: 0

MODIFICATION TO THE SAFETY INJECTION & RESIDUAL HEAT REMOVAL SYSTEM (O.C.) PER I.E. BULLETIN 79-14, PROBLEM 026

Summary:

This PC/M consisted in modifying the pipe supports on the safety injection and residual heat removal system (outside containment) to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 85-57

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 06/17/85

SUMMARY DATE: 10/02/85

REVISION: 0

STEAM GENERATOR BLOWDOWN SYSTEM FCV/BYPASS VALVE INTERLOCK

Summary:

This modification precludes the opening of the existing containment isolation valve with the downstream flow control valves (FCV) open during startup and normal system operation. The Steam Generator blowdown system had been experiencing hydraulic transients as a result of the opening of the isolation valves during startup of the system. The modification is accomplished by installing a limit switch on each flow control valve. A contact from each limit switch to the opening circuit of the respective bypass/main isolation valve provides a permissive interlock to prevent bypass/main isolation valve opening unless the respective flow control valve is closed.

Safety Evaluation:

This modification is non nuclear safety with QA-QC requirements. The probability/consequence of an accident previously evaluated in the FSAR will not be affected nor will the probability/consequence of a malfunction of equipment important to safety will be increased. There will be no margin of safety reduced as described in the Bases for any Technical Specifications. Thus, it can be concluded that this modification does not constitute any unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-40

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/28/85

SUMMARY DATE: 10/02/85

REVISION: 0

ENVIRONMENTAL SEALING OF PYCO RTD'S

Summary:

This modification involved the application of an epoxy sealing compound on the Pyco RTD head assemblies on the RCS hot and cold leg RTD's. This was performed to increase the watertight integrity of the subject RTD assemblies, thus increasing the qualified life of the subject RTD's to 40 years.

Safety Evaluation:

This PC/M does not involve an unreviewed safety question because it has no detrimental effect on any plant system or component important to safety and no margin of safety as defined in the Bases for any Technical Specifications was reduced.

PLANT CHANGE/MODIFICATION 84-24

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/07/85

SUMMARY DATE: 10/07/85

REVISION: 0

REPLACEMENT OF FLOW SWITCHES ON CHARCOAL FILTER BED FANS

Summary:

This change replaces the existing unqualified flow switches on the Emergency Containment Filters with qualified class IE thermal flow sensors from Fluid Components, Inc. (FCI).

Safety Evaluation:

This change is nuclear safety related because it affects the Emergency Containment Filters. It does not involve an unreviewed safety question because the new switches are fully qualified and independent train redundancy is maintained.

PLANT CHANGE/MODIFICATION 82-126

PC/M CLASSIFICATION:	NS
UNIT:	3
TURNED OVER DATE:	06/11/85
SUMMARY DATE:	10/21/85
REVISION:	0

MODIFICATION TO REACTOR COOLANT SYSTEM (I.C.) PER I.E. BULLETIN 79-14 (RTD-30)

Summary:

This change consisted in modifying the reactor coolant system pipe supports inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 83-138

PC/M CLASSIFICATION: NNS QA/QC

UNIT: 3 AND 4

TURNED OVER DATE: 04/26/85

SUMMARY DATE: 10/21/85

REVISION: 0

WATER SUPPRESSION FOR APPENDIX R MODIFICATIONS

Summary:

This PC/M installed preaction fire protection water suppression systems in: the Emergency Diesel Generator Bldg.; the Cable Riser Area in the Breezeway between the nuclear containments; Unit 3 Charging Pump and CCW areas; and Unit 4 Charging Pump and CCW areas. An isolation valve was also installed with each of the four above systems.

Safety Evaluation:

This PC/M is not safety related. However, so as not to interfere with existing safety related equipment in the case of seismic event, the systems installed by this PC/M will be supported to seismic Category I requirements. Provisions will also be made to protect electrical components from effects of water spray in the event of sparious actuation of the water suppression system. It is concluded that the probability of occurrence, or the consequences, of equipment malfunctions already evaluated by the FSAR is not increased or created by this PC/M. Therefore no unreviewed safety questions are outstanding.

PLANT CHANGE/MODIFICATION 83-178

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 06/11/85

SUMMARY DATE: 10/21/85

REVISION: 1

Negative Sequence Relay Timer - Unit

Summary:

This PC/M added an auxiliary relay that is used to initiate the starting and stopping of a new elapsed time device and provide a negative sequence alarm.

Safety Evaluation:

This PC/M is not safety related and the small cutout on the control board will have no adverse effect. Since no safety related systems are involved the probability of occurrance of an accident would not be greater.

PLANT CHANGE/MODIFICATION 83-190

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/07/85

SUMMARY DATE: 10/21/85

REVISION: 0

MAIN STEAM SENSING LINE FT-485 TUBING SUPPORT MODIFICATION

Summary:

The main steam sensing line to flow transmitter -485 has experienced excessive fatigue failure due to vibrations. This modification gave the tubing better support so that vibrations are dampened. The length of tubing still allows thermal expansion.

Safety Evaluation:

This changes does not involve an unreviewed safety question because:

a) The tubing is the same design, construction and rating as the original.

b) The tubing support does not interfere with any safety related function of the plant and is designed to original plant specifications and materials.

PLANT CHANGE/MODIFICATION 84-08

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 02/07/85

SUMMARY DATE: 10/21/85

REVISION: 0

MSIV/CV-4-204 POWER SUPPLY

Summary:

This PC/M removes the power supply for CV-4-204 (letdown isolation) from circuit 4D01-21 and connects it to circuit 4D01-31. This prevents adverse interaction between components powered from circuit 21.

Safety Evaluation:

This change is nuclear safety related but does not involve an unreviewed safety question because CV-4-204 is a containment isolation valve. The power supply shift is from a vital breaker to another vital breaker on the same division of DC power.

PLANT CHANGE/MODIFICATION CPWO 84-59 PC/M CLASSIFICATION: NNS QA-QC

UNIT: 3

TURNED OVER DATE: 05/03/85

SUMMARY DATE: 10/21/85

REVISION: 0

ADD STEEL GUARDS FOR H_2 MONITOR AE-3-6307

Summary:

This CPWO involved the addition of steel guards to the hydrogen monitors AE-6307 A & B located in the control room (racks 81 & 82).

Safety Evaluation:

The reason for this modification is to protect the meters against physical damage. Therefore, this CPWO is not nuclear safety related as it does not affect any safety related system or feature in the plant. Furthermore, it does not involve an unreviewed safety question as this modification does not affect, create, or increase the probability of occurrence of any accident or malfunction already addressed, or new, in the F.S.A.R.

PLANT CHANGE/MODIFICATION CPWO 84-60

PC/M CLASSIFICATION: NNS QA-QC

UNIT: 4

TURNED OVER DATE: 05/03/85

SUMMARY DATE: 10/21/85

REVISION: 0

ADD STEEL GUARDS FOR H_2 MONITOR AE-4-6307

Summary:

This CPWO involved the addition of steel guards to the hydrogen monitors AE-6307 A & B located in the control room (racks 81 & 82).

Safety Evaluation:

The reason for this modification is to protect the meters against physical damage. Therefore, this CPWO is not nuclear safety related as it does not affect any safety related system or feature in the plant. Furthermore, it does not involve an unreviewed safety question as this modification does not affect, create, or increase the probability of occurrence of any accident or malfunction already addressed, or new, in the F.S.A.R.

PLANT CHANGE/MODIFICATION 84-106

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/11/85

SUMMARY DATE: 10/21/85

REVISION: 0

MODIFICATION TO CHEMICAL AND VOLUME CONTROL SYSTEM (I.C.) PER I.E. BULLETIN 79-14 (CVCS-10)

Summary:

This change consisted in modifying the chemical and valve control system pipe supports inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 84-220

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 01/08/85

SUMMARY DATE: 10/21/85

REVISION: 0

SOURCE AND INTERMEDIATE RANGE DETECTOR WEEP HOLES

Summary:

This PC/M modified the source and intermediate range detectors by drilling two 1/2" holes in the base of the detectors. These weep holes will provide adequate drainage of borated water accumulated due to leaks in the cavity seal and seal liner.

Safety Evaluation:

This PC/M is safety related. The drilling of the weep holes does not affect the operation of the detectors. The probability of occurrence or consequences of an accident or equipment malfunction important to safety previously evaluated in the FSAR has not been increased.

PLANT CHANGE/MODIFICATION 85-21

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 02/04/85

SUMMARY DATE: 10/21/85

REVISION: 0

ENVIRONMENTAL SEALING OF PYCO RTD'S ON RCS HOT/COLD LEG LOOPS

Summary:

To increase the water tight integrity of RTD's assemblies, an approved sealing epoxy was applied to the head assembly. This modification effectively eliminated the potential moisture effect on measured process variable and increase qualified life of RTD assembly to 40 years.

Safety Evaluation:

The modification involves the application of a sealant to the terminal, gasket, and threads on the RCS hot and cold leg RTD's. The purpose of this modification is to hermetically seal the head assemblies and preclude moisture effects on measured variables in the event of LOCA. This modification does not involve unreviewed safety question, because this does not alter or effect any response time or any safety related equipment. For the above reasons, this changes does not increase the probability of occurrence or consequences of any accident, or equipment malfunctions described in the FSAR. By hermetically sealing the RTD head assemblies, possibility of adverse moisture effect on measurement is eliminated. Therefore, it improves the margin of safety and no potential for new accident is created.

PLANT CHANGE/MODIFICATION 80-42

PC/M CLASSIFICATION: NNS
UNIT: 3
TURNED OVER DATE: 11/04/85
SUMMARY DATE: 11/13/85
REVISION: 1

ADDITION OF STEAM GENERATOR ACCESS PLATFORM

Summary:

This modification provided access to the steam generator manway cover and to support a manway cover handling device and a manway cover storage compound.

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 82-15

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06-11-85

SUMMARY DATE: 11-13-85

REVISION: 0

MODIFICATION TO REACTOR COOLANT SYSTEM (I.C.) PER I.E. BULLETIN PROBLEM (PR-1)

Summary:

This change consisted in modifying the reactor coolant system pipe supports inside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

PLANT CHANGE/MODIFICATION 82-95

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06-11-85

SUMMARY DATE: 11-13-85

REVISION: 0

FISHER & PORTER TRANSMITTER REPLACEMENT

Summary:

This modification replaces non-qualified Fisher & Porter Pressure, Flow and Level Transmitters with qualified Rosemount Model Transmitters.

Safety Evaluation:

This change is nuclear safety related but does not involve an unreviewed safety question because the transmitters being replaced have inputs to the reactor protection system. The replacement transmitters are fully qualified for the expected environmental and seismic conditions expected.

PLANT CHANGE/MODIFICATION 84-51

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 05-11-85

SUMMARY DATE: 11-13-85

REVISION: 0

REACTOR CAVITY SUMP ACCESS DOOR REPLACEMENT

Summary:

This PC/M addressed the replacement of the existing Reactor Cavity Sump Access Door with one of sturdier construction. This door was needed to assure only authorized personnel would be allowed inside sump area.

Safety Evaluation:

This PC/M is non-nuclear safety related and does not involve an unreviewed safety question. However, since the door and its frame are installed inside the containment structure they have been designed to withstand the maximum seismic loading.

PLANT CHANGE/MODIFICATION 84-57

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 06/17/85

SUMMARY DATE: 11/13/85

REVISION: 0

INSTALL SWITCHYARD BREAKER 6B IN BAY 6 - UNIT 3

Summary:

This PC/M added a new 240 KV circuit breaker (6B) in Bay 6 of the switchyard.

Safety Evaluation:

This PC/M is not safety related. The added breaker improves operability and reliability of the plant. The probability of the occurrence of an accident previously evaluated in the FSAR will not increase. The probability of malfunction of equipment important to safety is not increased.

PLANT CHANGE/MODIFICATION 84-84

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 02/17/85

SUMMARY DATE: 11/13/85

REVISION: 0

TRANSFER OF CRDM COOLER FANS 3A & 3B

Summary:

This PC/M transferred the power to CRDM cooler fans from the non-vital to the vital busses.

Safety Evaluation:

The transfer of the CRDM fans to the vital busses improve the operability/reliability of the plant. The probability of occurrence of an accident previously evaluated in the FSAR will not increase.

PLANT CHANGE/MODIFICATION 84-92 PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 05/17/85

SUMMARY DATE: 11/13/85

REVISION: 0

AIR DRYER FOR STEAM JET AIR EJECTOR RAD MONITOR

Summary:

This PC/M package installed chillers on the discharge of the SJAE. The purpose of the chillers is to remove moisture from the SJAE exhaust in order to provide a dry sample to the SJAE Sping-4 Radiation Monitor. This design will be installed temporarily until a permanent system can be developed.

Safety Evaluation:

This is a non-safety related design change. The failure of the chiller system will not affect safety-related systems, components or structures. The failure of this equipment will not increase the probability of occurrence of a nuclear-related accident nor will it degrade the performance of any system components mitigating an accident. This system will be upgraded to a permanent configuration by a future PC/M.

PLANT CHANGE/MODIFICATION 84-132

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05/28/85

SUMMARY DATE: 11/13/85

REVISION: 1

VITAL POWER SUPPLY FOR PC-600, 601 CONTROL RELAYS

Summary:

This PC/M provided separate and redundant class IE power supplies to the control relays for the RHR pump discharge pressure controllers.

Safety Evaluation:

This PC/M is safety related, the new class IE power supplies will improve the reliability of the plant. The probability of the occurrence of an accident previously evaluated in the FSAR will not increase or the consequences of an accident previously evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 84-133

PC/M CLASSIFICATION:	NS
UNIT:	4
TURNED OVER DATE:	02/08/85
SUMMARY DATE:	11/13/85
REVISION:	0

VITAL POWER SUPPLY FOR PC-600, 601 CONTROL RELAYS

Summary:

This PC/M provided separate and redundant class IE power supplies to the control relays for the RHR pump discharge pressure controllers.

Safety Evaluation:

This PC/M is safety related, the new class IE power supplies will improve the reliability of the plant. The probability of the occurrence of an accident previously evaluated in the FSAR will not increase or the consequences of an accident previously evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 84-202

PC/M CLASSIFICATION: NS

UNIT: 3 AND 4

TURNED OVER DATE: 09-30-85

SUMMARY DATE: 11-13-85

REVISION: 1

MCC CHANGES & CONSTANT VOLTAGE TRANSFORMERS FOR INVERTER REPLACEMENT

Summary:

This PC/M provided a regulated 120V AC backup supply for each of the normally operating vital inverters.

Safety Evaluation:

This PC/M is safety related. The probability of occurrence of an accident previously evaluated in the FSAR will not be increased since it does not change the function of any plant system, and the loading of the vital busses is not increased. Also the consequence of an accident previously evaluated in the FSAR will not be affected.

PLANT CHANGE/MODIFICATION 85-115

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 11/04/85

SUMMARY DATE: 11/13/85

REVISION: 0

REACTOR TRIP BREAKER AUTO SHUNT TRIP

Summary:

This PC/M relocated to a subpanel for easy accessibility the auto shunt trip for the reactor trip breaker.

Safety Evaluation:

This PC/M is safety related. It does not involve an unreviewed safety question since this relocation of equipment does not adversely affect the seismic qualification of the equipment and no functional changes have been made in the breaker control circuit.

PLANT CHANGE/MODIFICATION 84-127

PC/M CLASSIFICATION: NS

UNIT: 3 & 4

TURNED OVER DATE: 09/30/85

SUMMARY DATE: 11/13/85

REVISION: 0

ADDITION OF DIESEL GENERATOR COOLING SYSTEM INSTRUMENTATION

Summary:

This PC/M replaced the lube oil cooler, oil inlet and outlet temperature indicators for qualified ones; new thermowells were installed at the radiator cooling water inlet and outlet pipelines. Added a thermocouple to the turbocharger exhaust inlet manifold and a new crankcase pressure manometer.

Safety Evaluation:

This PC/M is safety related, no safety characteristics will be changed and the probability of an accident would not be greater because the additional inst. has no interconnection with any other S.R. system. The consequences of an accident previously evaluated in the FSAR has not increased.

PLANT CHANGE/MODIFICATION 82-296

PC/M CLASSIFICATION: NNS

UNIT: 3 & 4

TURNED OVER DATE: 01/11/85

SUMMARY DATE: 12/13/85

REVISION: 0

STANDBY STEAM GENERATOR FEEDPUMP

Summary:

This PC/M requires the addition of two non-safety related feedwater pumps for standby operation. They are to be used for normal plant startup and shutdown.

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 question.

PLANT CHANGE/MODIFICATION 83-65

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06/14/85

SUMMARY DATE: 12/13/85

REVISION: 0

RHR & HHS/VALVE POSITION INDICATION

Summary:

This modification disconnected the white light indication on VPB for valves in the RHR and HHSI system. It reconnected the indicators to a separate vital 120V AC source and changed lens covers to amber. This provides a continuous valve position indication even with the breakers racked out.

Safety Evaluation:

This change is nuclear safety related but does not involve an unreviewed safety question, because the modification does not affect the valve circuits but provides improved valve position indication.

PLANT CHANGE/MODIFICATION 83-69

PC/M CLASSIFICATION: NNS-QA/QC

UNIT: 3

TURNED OVER DATE: 05/01/85

SUMMARY DATE: 12/13/85

REVISION: 0

REACTOR HEAD SHIELDING

Summary:

The subject modification provided a permanent circular beam structure upon which temporary lead shielding (3/4" thick lead wool blankets) is to be hung. This PC/M provided the structure, hardware, lead blankets and stainless steel storage bin.

Safety Evaluation:

This non-nuclear safety realted PC/M provides shielding around the circumference of the reactor vessel head from the top of the lead flange to an elevation 7'0" above the flange. No increase in the probability for an accident results. No new type of accident is created. Seismic design requirements meet or exceed those for the head lift rig. No reduction in the margin of safety defined in the basis for any Tech. Spec. results.

PLANT CHANGE/MODIFICATION 83-115

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 10/08/85

SUMMARY DATE: 12/13/85

REVISION: 0

PRESSURIZER HEATER AMMETERS

Summary:

This PC/M added an ammeter, phase selector switch, and an additional current transformer to each of the three pressurizer heater group breakers. This will provide a way to determine if one of the heater groups is not functioning properly.

Safety Evaluation:

The pressurizer heater ammeters and CT's installed by this PC/M are not safety related nor do they serve as an input to a safety related function. The probability of occurrence or the consequences of a design basis or malfunction of equipment important to the safety of the plant has not been increased. Therefore, it can be concluded that this PC/M does not pose an unreviewed safety question.

PLANT CHANGE/MODIFICATION 83-136

PC/M CLASSIFICATION: NNS/QA-QC

UNIT: 3

TURNED OVER DATE: 05/28/85

SUMMARY DATE: 12/13/85

REVISION: 0

WATER SUPPRESSION SYSTEM FOR APPENDIX "R" MODIFICATION

Summary:

This modification provided for the installation of water suppression systems to satisfy licensing commitments associated with Appendix "R" requirements. Areas affected were the charging pump room and the component cooling water equipment area.

Safety Evaluation:

This modification is non-nuclear safety related with QA-QC requirements. There is no safety question involved, since there is no probability/consequences of an accident/malfunction to any equipment important to safety. These modifications relate to Technical Specifications 3.14 and 4.15. The margins 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.

PLANT CHANGE/MODIFICATION 83-137

PC/M CLASSIFICATION: NNS/QA-QC

UNIT: 4

TURNED OVER DATE: 04/26/85

SUMMARY DATE: 12/13/85

REVISION: 0

WATER SUPPRESSION SYSTEM FOR APPENDIX "R" MODIFICATION

Summary:

This modification provided for the installation of water suppression systems to satisfy licensing commitments associated with Appendix "R" requirements. Areas affected were the charging pump room and the component cooling water equipment area.

Safety Evaluation:

This modification is non-nuclear safety related with QA-QC requirements. There is no safety question involved, since there is no probability/consequences of an accident/malfunction to any equipment important to safety. These modifications relate to Technical Specifications 3.14 and 4.15. The margins 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.

PLANT CHANGE/MODIFICATION 84-129

PC/M CLASSIFICATION: NS

UNIT: 3 & 4

TURNED OVER DATE: 08/13/85

SUMMARY DATE: 12/13/85

REVISION: 1

GLAND FLANGE REPAIRS FOR 3/4" ROCKWELL EDWARDS VALVE (T58)

Summary:

This modification consisted in fabricating and installing temporary "strongback" plates which will replace original damaged valve gland flanges on all of the 3/4" Rockwell Edwards valves T58. The following valves have been modified: Unit 3 Valves: 538, 564A, 304G, 954C, 304C, 954A; Unit 4 Valve: 514. This PC/M will remain open until the next refueling outages and will be used to work other similar valves should there be other leakage. Subsequent report will follow documenting said changes.

Safety Evaluation:

This temporary modification does not have a direct effect on any plant feature necessary to ensure the integrity of the reactor coolant system pressure boundary, the capability to safely shut down the reactor or the capability to prevent or mitigate the consequences of accidents which could result in exposures comparable to the guideline exposures described in 10 CFR 100.

In addition, this temporary change does not increase the probability of an accident or malfunction of equipment important to safety nor does it create the possibility of an accident or malfunction not previously evaluated in the FSAR. Furthermore, it does not reduce the margin of safety as defined in the basis for any Technical Specification.

PLANT CHANGE/MODIFICATION 84-168

PC/M CLASSIFICATION: NSR

UNIT: 3 & 4

TURNED OVER DATE: 10/04/85

SUMMARY DATE: 12/13/85

REVISION: 0

LOCKOUT OF DIESEL GENERATOR POWER TO THE MOTOR DRIVEN FIRE PUMP

Summary:

This PC/M modified the existing power supply so that the pump will be locked out upon presence of any combination of an SIS and undervoltage of the two units.

Safety Evaluation:

This PC/M is safety related. The motor driven fire pump is essentially stand-by equipment required for fire suppression, the probability of occurrence of an accident previously evaluated in the FSAR is not affected by this PC/M.

PLANT CHANGE/MODIFICATION 82-296

PC/M CLASSIFICATION: NNS

UNIT: 3 & 4

TURNED OVER DATE: 04/18/85

SUMMARY DATE: 12/18/85

REVISION: 2

STANDBY STEAM GENERATOR FEEDPUMP

Summary:

This PC/M requires the addition of two non-safety related feedwater pumps for standby operation. They are to be used for normal plant startup and shutdown.

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 question.

PLANT CHANGE/MODIFICATION 84-63 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 12-19-85

SUMMARY DATE: 02-05-86

REVISION: 0

REPLACEMENT OF DIESEL GENERATOR CONTROL ROOM ISOLATION SWITCH

Summary:

This PC/M changed the selector switch on cabinet 3C12 from (Local-Off-Normal) to (Local-Normal-Off) to avoid D/G trips.

Safety Evaluation:

This PC/M is safety related since the new SB-9 G.E. switch is fully qualified and no functional modifications in the control circuit are involved, the probability of occurrence of an accident previously evaluated in the FSAR is unchanged, and the consequences of an accident previously evalauted in the FSAR are not adversely affected.

PLANT CHANGE/MODIFICATION 84-64 PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 12-17-85

SUMMARY DATE: 02-05-86

REVISION: 0

REPLACEMENT OF DIESEL GENERATOR CONTROL ROOM ISOLATION SWITCH

Summary:

This PC/M changed the selector switch on cabinet 3C12 from (Local-Off-Normal) to (Local-Normal-Off) to avoid D/G trips.

Safety Evaluation:

This PC/M is safety related since the new SB-9 G.E. switch is fully qualified and no functional modifications in the control circuit are involved, the probability of occurrence of an accident previously evaluated in the FSAR is unchanged, and the consequences of an accident previously evalauted in the FSAR are not adversely affected.

PLANT CHANGE/MODIFICATION 84-157

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 12-19-85

SUMMARY DATE: 02-05-86

REVISION: 0

REPLACEMENT OF TYPE CFD DIFFERENTIAL RELAY IN "A" EMERENCY DIESEL GENERATOR

Summary:

This PC/M replaced the existing CFD current differential relays with IJD relays.

Safety Evaluation:

This PC/M is safety related. It does not involve an unreviewed safety question since; it improved the vibrational fragility of the diesel generator differential relay circuit, the replacement relays have been seismically tested. It does not adversely affect operation of the EDG, it does not change the margin of safety as defined in the Basis for the D.G Tech. Spec. 3.7.

PLANT CHANGE/MODIFICATION 84-110

PC/M CLASSIFICATION: NNS

UNIT: 3 & 4

TURNED OVER DATE: 01-15-86

SUMMARY DATE: 02-11-86

REVISION: 0

HEALTH PHYSICS CONTROL FACILITY

Summary:

This modification consisted in providing a 60' x 150' pre-engineered metal building for the Health Physics Department to consolidate their functions into one location. This modification covers the building structure and all interior work. The site preparation, electrical, water, fire water, sewage and service air tie-ins are covered under PC/M 84-109.

Safety Evaluation:

Based on the above, it can be concluded that 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 accident or malfunction different than those previously evaluated in the FSAR. Also, there are no changes to the Technical Specifications of the plant. Therefore, it can be concluded that the new HP Control Facility does not pose any unreviewed safety question.

PLANT CHANGE/MODIFICATION 84-148

PC/M CLASSIFICATION:	NS
UNIT:	4
TURNED OVER DATE:	12-19-85
SUMMARY DATE:	02-11-86
REVISION:	0

BORIC ACID TRANSFER PUMP DRAIN VALVE

Summary:

This PC/M installed a drain valve on the casing of the Goulds model 3196 size 1 x 1-1/2 - 8 Boric Acid Transfer Pump (BATP). The drain valve is intended to facilitate maintenance of the pumps.

The PC/M consisted of a short 1/2" pipe nipple, drain valve, and additional nipple piece.

Safety Evaluation:

The old BATP pumps had drain valves, the new pumps installed under PC/M 80-83 do not. This design restored the boric acid transfer system to its previous condition by installing a drain valve on the new pumps. This PC/M therefore does not change the probability of occurrence or consequences of an accident previously evaluated in the FSAR. It does not increase the probability or consequences of a malfunction previously evaluated in the FSAR. It does not create the possibility of an accident of different type or malfunction of a different type than those previously analyzed in the FSAR.

PLANT CHANGE/MODIFICATION 85-08

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 12-23-85

SUMMARY DATE: 02-11-86

REVISION: 0

MODIFICATION TO COVERED WALKWAY FROM AUXILIARY BUILDING TO CONTAINMENT ENTRANCE

Summary:

This PC/M addresses the extension of the covered walkway area, originally installed in accordance with PC/M 80-166, along the existing north-east roof line to the Unit 4 containment wall as well as the addition of side panels on the south and west faces of the 33' level covered area. Also included in this PC/M is the installation of additional lighting necessary due to the extension of the covered area.

The covered walkway from the auxiliary building to the Unit 4 containment entrance is a structural steel and aluminum roofing system intended to provide dry passage to the containment personnel hatch in order to preclude the spread of contamination by rainwater.

Safety Evaluation:

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 the installation of the additional covered walkway and lighting system does not pose an unreviewed safety question pursuant to 10 CFR 50.59.

PLANT CHANGE/MODIFICATION 85-58

PC/M CLASSIFICATION: NNS QA-OC

UNIT: 4

TURNED OVER DATE: 11-29-85

SUMMARY DATE: 02-11-86

REVISION: 0

STEAM GENERATOR BLOWDOWN SYSTEM FCV/BYPASS VALVES INTERLOCK

Summary:

This PC/M provided the design of the Steam Generator (S/G) Blowdown System Flow Control Valve/Bypass Valve Interlock for Unit 4. The modification to the existing system provides a position indicating limit switch, mounted on the flow control valve. A contact from the limit switch will interlock the open circuit of the bypass/main isolation valve. This contact indicates a closed position for the flow control valve and precludes opening of the bypass/main isolation valve unless the flow control valve is fully closed.

Safety Evaluation:

This modification is Non-Nuclear Safety with no Unreviewed Safety Questions, since there is no change in the probability or consequence of an accident on equipment malfunction as previously evaluated in the FSAR. With respect to the margin of safety as defined in the Basis for any Technical Specification, no margin will be reduced.

PLANT CHANGE/MODIFICATION 85-113

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 11/29/85

SUMMARY DATE: 02/11/86

REVISION: 0

STEAM GENERATOR BLOWDOWN ISOLATION VALVE TIME DELAY RELAY MODIFICATION

Summary:

This PC/M provides the design to utilize the existing non-Class 1E time delay relays, identified by NCR 342-85 dated June 18, 1985, common in the Steam Generator Blowdown isolation valves safety related control circuits. The safety related function of these valves is to close in response to Containment Isolation - Phase A or Auxiliary Feedwater start signals. The modification rewires the control power feed to the isolation valve downstream of normally closed, open on trip, isolation contacts. The existing time delays are seismically installed under PC/M 78-102B. Design for the Steam Generator Blowdown Isolation Valve Time Delay Relay Modifications was done to meet the requirements of the design inputs and project standards using applicable design methods.

Safety Evaluation:

This modification is Nuclear Safety with no Unreviewed Safety Questions, since there is no change in the probability or consequence of an accident on equipment malfunction as previously evaluated in the FSAR. With respect to the margin of safety as defined in the Basis for any Technical Specification, no margin will be reduced.

PLANT CHANGE/MODIFICATION 82-123

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06-24-85

SUMMARY DATE: 02-25-86

REVISION: 0

ACCUMULATOR LEVEL TRANSMITTER MODIFICATION

Summary:

This modification consisted in the replacement of existing accumulator level transmitters with qualified Rosemount 1153 Series D Differential Pressure Level Transmitter. It also fitted this new transmitter with qualified Conax Electric Conductor Seal Assemblies, changed existing level indicators scales (range 2000 to 9000 gal.) and modified the power supply to the new transmitter to provide for channel separation. This PC/M was implemented to meet the intent of Regulatory Guide 1.97 by installing level transmitters which are qualified for the environment, in accordance with IEEE-323-1974 and IEEE-344-1975.

Safety Evaluation:

This change is Nuclear Safety Related. However, it does not involve an unreviewed safety question. The installation of Conax Electric Seal Assemblies will maintain the Environmental Qualification of the Rosemount 1153 Series D pressure transmitter. No system characteristic will be changed and the probability of occurrence of an accident or equipment malfunction, new or already evaluated in the FSAR is not created or increased. This modification does not decrease the design margins of the system, change the operating function or conditions or affect other safety related equipment.

PLANT CHANGE/MODIFICATION 84-151

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 10-28-85

SUMMARY DATE: 02-25-86

REVISION: 0

TPCW/CCW CATHODIC PROTECTION UNITS 3 & 4

Summary:

This PC/M installed the electrical portion of the cathodic protection system for the CCW & TPCW.

Safety Evaluation:

This PC/M is not safety related and has no effect on S.R. systems. It has no effect on the probability of occurrence of an accident already evaluated in the FSAR. No equipment important to safety would be made more likely to malfunction by the installation of the system.

PLANT CHANGE/MODIFICATION 84-152

PC/M CLASSIFICATION:	NNS
UNIT:	4
TURNED OVER DATE:	10-28-85
SUMMARY DATE:	02-25-86
REVISION:	0

TPCW/CCW CATHODIC PROTECTION UNITS 3 & 4

Summary:

This PC/M installed the electrical portion of the cathodic protection system for the CCW & TPCW.

Safety Evaluation:

This PC/M is not safety related and has no effect on S.R. systems. It has no effect on the probability of occurrence of an accident already evaluated in the FSAR. No equipment important to safety would be made more likely to malfunction by the installation of the system.

PLANT CHANGE/MODIFICATION 82-163

PC/M CLASSIFICATION: NNS-OA/OC

UNIT: 4

TURNED OVER DATE: 12-28-84

SUMMARY DATE: 02-25-86

REVISION: 1

EXISTING FIRE TANK SYSTEM MODIFICATIONS

Summary:

This PC/M will modify the existing 500,000 gallon raw water storage tank so that 300,000 gallons are dedicated for fire protection. This will include relocation of raw water lines to a higher inlet, new tank nozzles, valves, and additional component supports, tying into existing and new piping associated with a 750,000 gallon tank and diesel pump. This is part of NRC requirements for a total of 600,000 gallons dedicated for fire protection from two separate, redundant sources.

Safety Evaluation:

Modifications to the existing 500,000 gallon tank and cross connecting the raw water tanks is non-nuclear safety related since the tanks, fire pumps, and raw water pumps do not perform a nuclear safety function. The modified equipment is located away from nuclear safety related structures, systems, and components. 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 or different from those previously evaluated, has not been increased. Therefore, it can be concluded that this PC/M does not pose an unreviewed safety question.

PLANT CHANGE/MODIFICATION 84-27

PC/M CLASSIFICATION: NNS

UNIT: 3/4

TURNED OVER DATE: 1-21-85

SUMMARY DATE: 02-28-86

REVISION: 1

DEDICATED FIRE PROTECTION SYSTEM MODIFICATIONS

Summary:

This change provided separation of Fire Protection System from Service Water System Services and the removal of non-fire protection related services from the fire loop. This change was reviewed with respect to FSAR; the FPL Fire Protection Review Report; 10 CFR 50, Appendix R, Section III.A; L-84-30 and associated SER; PTP Tech. Specs.; and Scope Change 354-01.

Safety Evaluation:

The probability of occurrence of an accident previously evaluated in the FSAR will not be increased because these changes do not alter the facility with respect to the RCS, Main Steam System, Fuel Handling, Storage, or any other system or component associated with plant safety.

PLANT CHANGE/MODIFICATION 84-147

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 10-10-85

SUMMARY DATE: 02-28-86

REVISION: 0

BORIC ACID TRANSFER PUMP DRAIN VALVE

Summary:

This modification consisted in installing a drain valve on the casing of the Goulds Model 3196 size 1 x 1-1/2 - 8 Boric Acid Transfer Pumps, to facilitate maintenance of the pumps.

Safety Evaluation:

This PC/M does not change the probability of occurrence or consequences of an accident previously evaluated in the FSAR. It does not increase the probability or consequences of a malfunction previously evaluated in the FSAR. It does not create the possibility of an accident of different type or malfunction of a different type than those previously anlayzed in the FSAR.

PLANT CHANGE/MODIFICATION 85-103

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 06-22-85

SUMMARY DATE: 02-28-86

REVISION: 0

NIS INPUT TO TURBINE RUNBACK - CHANGE TO 2/4 LOGIC

Summary:

This modification changed the Unit 4 - Logic for initiating a turbine runback to 70% of turbine load caused by a negative flux rate input (NIS signal) from 1 out of 4 to 2 out of 4 on an interim basis. The permanent modification will be completed under PC/M 84-211 and will revert to a 1 out of 4 channel logic.

Safety Evaluation:

This modification is Nuclear Safety Related with the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not changed relative to this change. A possibility for an accident or malfunction of a different type than has been previously evaluated in safety analysis reports is not created due to the change to 2/4 logic. The margin of safety as defined in the basis for any technical specifications is not reduced. It, therefore, can be stated that this proposed modification has not reduced any margin of safety since the generic methodology, previously approved by NRC, demonstrated the equal ability of turbine runback to be acutated on one-out-of-four logic or two-out-of-four logic for the multiple dropped rod event.

PLANT CHANGE/MODIFICATION 82-236

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 06-13-85

SUMMARY DATE: 03-13-86

REVISION: 0

CONTAINMENT PURGE VALVE SEATS & HUB SEAL REPLACEMENT

Summary:

The purpose of this PC/M was to replace the existing purge valve seats and hub seals made of nitrile rubber with those made of ethylene propylene terpolymer (EPT). Also, the replacement of the valve cover gaskets, bearings and packing were a part of this PC/M.

Safety Evaluation:

Nuclear Safety Related, the valves are used to mitigate the consequences of an accident (i.e., LOCA).

This modification does not involve an unreviewed safety question because it would not decrease any margin of safety discussed in the Technical Specifications.

PLANT CHANGE/MODIFICATION 83-64

PC/M CLASSIFICATION: NNS

UNIT: 4

TURNED OVER DATE: 02-22-86

SUMMARY DATE: 03-13-86

REVISION: 0

IMPROVED FLOOR DRAINS FOR CONTAINMENT SPRAY PUMP ROOM

Summary:

This PC/M modifies the drains in the Containment Spray Pump Room by bypassing the flow restriction to provide an easier flow path to the drain header. The Containment Spray Pump Room floor drain did not operate in an acceptable fashion and on several occasions the drain has overflowed. This causes the water to backup into the Containment Spray Pump Room and sometimes into the Auxiliary Building hallway.

Safety Evaluation:

With respect to the probability of occurrence of an accident previously evaluated in the FSAR: This modification is part of the plant design improvement program and does not affect the safety of safety related equipment. Therefore, this PC/M does not involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION CPWO 84-143 PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 03-07-86

SUMMARY DATE: 03-13-86

REVISION: 0

REPLACEMENT OF G.E. 12 HFA51A42F RELAYS

Summary:

This modification consisted in replacing G.E. HFA Auxiliary Relays experiencing failures attributed to coil spool material deficiencies as per NRC I.E. Bulletin 84-02 with G.E. Century series type HFA relays.

Safety Evaluation:

On the basis of a one for one exchagne of non-Century series HFA relays for Century series HFA relays the following statements are justified:

a. The probability of occurrence or the consequences of design basis accidents or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

b. The possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR is not created.

c. The margin of safety as defined in the basis for any Technical Specification is not reduced.

Therefore, no unreviewed safety question as defined in 10 CFR 50.59 is involved.

PLANT CHANGE/MODIFICATION 85-46

PC/M CLASSIFICATION: NNS QA/QC

UNIT: 3

TURNED OVER DATE: 04-11-85

SUMMARY DATE: 03-13-86

REVISION: 0

SPENT FUEL PIT TOOL STORAGE RACK MODIFICATION

Summary:

Since the installation of the spent fuel pit tool storage rack PC/M 79-102, a new tool has been obtained for use in the spent fuel pool. The Westinghouse Portable Rod Cluster Control (RCC) change tool is both heavier and longer than the tools the rack was designed to support.

The existing steel storage rack was extended to the east to provide storage for the new tool. This extension is located at the north edge of the spent fuel pit at the 58 ft. elevation. Approximately nine feet of the new RCC change tool extends outside the spent fuel pit in order to hang on this new rack.

Safety Evaluation:

The additional tool rack does not perform a safety function or provide protection for safety related systems or equipment.

As discussed in the design analysis, the Spent Fuel Pit structure and liner plate will not be adversely affected by this modification. No other safety related components interact with the additional tool rack. No additional loads will be imposed on the existing tool storage rack.

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 the modification of the Spent Fuel Pit tool storage rack does not pose an unreviewed safety question pursuant to 10 CFR 50.59.

PLANT CHANGE/MODIFICATION 86-08

PC/M CLASSIFICATION: NNS QA/QC

UNIT: 4

TURNED OVER DATE: 03-07-86

SUMMARY DATE: 03-13-86

REVISION: 0

SPENT FUEL PIT CAMERA MONITORS

Summary:

This modification consisted in installing two surveillance cameras, furnished by the International Atomic Energy Agency (IAEA) on the walls of the spent fuel pit building. These cameras are required as a result of the United States participation in the Nuclear Non-Proliferation Treaty.

Safety Evaluation:

The probability of occurrence of an analyzed accident remains in unaffected since no equipment or components important to safety are affected and no fuel damage will occur as a result of this modification. The only potential new accident created is the failure of the camera and it does not perform any safety function. Finally the margin of safety as defined in the Technical Specifications has not been reduced because no safety functions controlled by the Technical Specifications are affected. Based on the above it can be concluded that an unreviewed safety question does not exist.

PLANT CHANGE/MODIFICATION CPWO 86-013 PC/M CLASSIFICATION: NS

UNIT: 3/4

TURNED OVER DATE: 02-24-86

SUMMARY DATE: 03-13-86

REVISION: 0

AFW MECHANICAL OVERSPEED TRIP DEVICE

Summary:

This CPWO covered the replacement of the tappet and ball assembly in "A" Auxiliary Feedwater Pump Turbine with an upgraded assembly supplied by Terry Turbine as a spare part for the turbines. The failed part number PC #047111 has been replaced by Terry part number PC #131654C01 and is identical in fit, form and function.

Safety Evaluation:

The replacement of the tappet and ball holder assembly for "A" Auxiliary Feedwater Pump Turbine does not involve an unresolved safety question because the probability of occurrence or the consequences of design basis accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased, and 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 84-205

PC/M CLASSIFICATION: NS

UNIT: 3 AND 4

TURNED OVER DATE: 11-22-85

SUMMARY DATE: 04-07-86

REVISION: 1

RACEWAYS FOR INVERTER REPLACEMENT

Summary:

This PC/M installed conduits, junction boxes and trays as well as foundation for constant voltage transformers (CVT), transfer switches, and synchronization switches for PC/M 83-117 inverter replacement.

Safety Evaluation:

This PC/M is safety related. All raceways and equipment were seismically supported. The probability of occurrence of an accident previously evaluated in the FSAR will not be increased. There is no possibility that an accident will be created which is of a diferent type than already evaluated in the FSAR.

PLANT CHANGE/MODIFICATION 85-79 PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 09-10-85

SUMMARY DATE: 04-07-86

REVISION: 0

CATHODIC PROTECTION CONTROL CABINET RHEOSTAT REMOVAL

Summary:

This PC/M replaced the Rheostat in the intake, condenser and containment cathodic protection control cabinets with jumpers to improve the cathodic protection system reliability.

Safety Evaluation:

This modification, therefore, does not increase the probability of occurrence or the consequences of a design basis accident or the malfunction of equipment important to safety previously evaluated in the FSAR. The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR will not be created. Cathodic protection is not addressed in the Technical Specifications and, therefore, this change will not reduce the margin of safety as defined in the basis for a Technical Specification.

This PC/M is thus, not safety related and is deemed not to involve an unreviewed safety question.

PLANT CHANGE/MODIFICATION 85-111

PC/M CLASSIFICATION: NNS

UNIT: 3

TURNED OVER DATE: 11-13-85

SUMMARY DATE: 04-07-86

REVISION: 0

S/G BLOWDOWN FCV TIMING MODIFICATION

Summary:

This modification provided a needle valve on each flow control valve (FCV) on the S/G Blowdown to slow down the control air delivered to the valve actuators, thereby slowing the control valve response to open. Thus providing a more reliable system operation and availability by mitigating system transients resulting in piping and support damage.

Safety Evaluation:

This PC/M is non-nuclear safety with no unreviewed safety question since the probability/consequences of an accident/malfunction was not increased nor was margin of safety reduced, as described in the Basis for any Technical Specification.

PLANT CHANGE/MODIFICATION 80-83

PC/M CLASSIFICATION: NS

UNIT: 3 AND 4

TURNED OVER DATE: 05-28-85

SUMMARY DATE: 04-22-86

REVISION: 2

BORIC ACID TRANSFER PUMP REPLACEMENT

Summary:

The boric acid transfer pumps are being replaced in an attempt to increase reliability. The old pumps were susceptible to seal and bearing failures. The new pumps employ a different cooling design. At the present time, only the 4A pump has been replaced with replacement of the others planned during the summer of '83.

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 83-34

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 11-13-85

SUMMARY DATE: 06-16-86

REVISION: 1

ADDITION OF RWST LEVEL INDICATION

Summary:

This modification removes the existing pneumatic RWST Level Indication and replaces it with qualified redundant electronic loops.

Safety Evaluation:

This change is Nuclear Safety Related because it affects the monitoring of the RWST Level in the Safety Injection System. It does not involve an unreviewed safety question because the new equipment is of higher quality than the old equipment.

PLANT CHANGE/MODIFICATION CPWO 84-61

PC/M CLASSIFICATION: NS

UNIT: 4

TURNED OVER DATE: 02-11-86

SUMMARY DATE: 06-16-86

REVISION: 0

REFUELING CRANE LIMIT SWITCH

Summary:

This modification consisted in permanently removing limit switch 22 by use of a jumper. The limit switch did not serve any further use, since the removal of the West reactor head quide stud was done. This limit switch was used to protect the underwater T.V. mast from hitting the quide stud during refueling.

Safety Evaluation:

The refueling crane electrical system is not nuclear safety related and does not perform any nuclear safety function. Therefore, this CPWO is not nuclear safety related as it does not affect any safety related system or features in the plant. Furthermore, it does not involve any unreviewed safety question as this modification does not affect, create, or increase any accident/malfunction already addressed or new in the FSAR. Basis for Technical Specifications remain unchanged.

PLANT CHANGE/MODIFICATION 84-139

PC/M CLASSIFICATION: NS

UNIT: 3 & 4

TURNED OVER DATE: 09-20-85

SUMMARY DATE: 06-16-86

REVISION: 0

BORIC ACID BATCHING TANK AGITATOR REPLACEMENT

Summary:

This PC/M replaced the existing boric acid batching tank agitator (Model FGBS-1) with a more durable model (GS-2).

Safety Evaluation:

This PC/M is safety related. The failure of the new agitator will not hinder the functional ability of the batching tank to mitigate the consequences of an accident. The probability of occurrence or the consequences of a design accident or malfunction of equipment important to the plant, previously evaluated in the FSAR, has not been increased.

PLANT CHANGE/MODIFICATION	**84-197**	PC/M CLASSIFICATION:	NS
		UNIT:	4
		TURNED OVER DATE:	03-21-86
		SUMMARY DATE:	06-16-86
		REVISION:	0

BACKUP BOTTLED NITROGEN SUPPLY TO MSIV'S

Summary:

This design package provided for modifications to the Unit 4 Main Steam Isolation Valves (MSIV's) instrument air supply system to provide a seismically designed bottled nitrogen supplemental backup to the existing air reserve tanks. The backup nitrogen system is designed to enhance the existing system by providing an additional reserve of nitrogen to keep the MSIVs closed for at least 30 minutes in case of a failure of the instrument air supply.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety questions since, the probability of occurrence of an accident previously evaluated in the FSAR will not be increased nor will any consequence of equipment malfunction increase. The margin of safety as defined in the Bases for the Technical Specifications will not be reduced.

PLANT CHANGE/MODIFICATION 84-192

PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 05-08-85

SUMMARY DATE: 06-13-86

REVISION: 0

MODIFICATION TO CCW SYSTEM (O.C.) PER I.E. BULLETIN 79-14, CCW-47

Summary:

This PC/M modified the pipe supports in the Component Cooling Water outside containment to comply with NRC I.E. Bulletin 79-14.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original piping system design documents have been met and no accident or malfunction probability increased.

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, 1985, to June 30, 1986.

Unit 3

In accordance with Operating Procedure 3-OP-041.4, Overpressure Mitigating System, PORV 455C and/or 456 were tested for operability on December 1, 1985, December 3, 1985, January 8, 1986, January 12, 1986, January 25, 1986, March 5, 1986, March 9, 1986, and on March 28, 1986. One or both valves were satisfactorily tested on the above dates. Additional valve actuations are listed below.

July 7, 1985	PORV 455C and 456 were opened as per OP-1004.4, Overpressure Mitigating System - Functional Test of N_2 Backup. Both PORVs stroked satisfactorily.
October 21, 1985	PORV 455C prematurely opened twice during a cooldown of the reactor coolant system. PORV operation was satisfactory.
October 24, 1985	PORV 456 was cycled to demonstrate its operability following maintenance work. The valve operation was satisfactory during post maintenance testing.
November 1, 1985	PORV 455C and 456 were opened as per OP-1001.1, Filling and Venting The Reactor Coolant System. Both PORVs stroked satisfactorily.
January 12, 1986	PORV 455C cycled open due to a reactor coolant system pressure spike during low pressure operation. PORV operation was satisfactory.
April 3, 1986	PORV 455C cycled open due to a reactor coolant system pressure spike. PORV operation was satisfactory.

Unit 4

In accordance with Operating Procedure 4-OP-041.4, Overpressure Mitigating System, PORV 455C and/or 456 were tested for operability on November 24, 1985, November 30, 1985 and on April 19, 1986. One or both valves were satisfactorily tested on the above dates. Additional valve actuations are listed below.

August 24, 1985	PORV 455C and 456 were opened as per OP-1001.1, Filling and Venting the Reactor Coolant System. Both PORVs stroked satisfactorily.
January 10, 1986	PORV 455C was cycled due to the operation of PC-444J, auto-manual setpoint station, on the control room console. PORV operation was satisfactory.
March 9, 1986	PORV 455C was cycled as a part of the startup test for the alternate shutdown panel. PORV operation was satisfactory.
April 5, 1986	PORV 456 was cycled as per 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV operation was satisfactory.
April 6, 1986	PORV 455C was cycled as per 4-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. PORV 455C failed to open when test signal was generated. Maintenance was performed on the valve and the valve was subsequently tested on April 10, 1986 as per 4-OSP-041.4. PORV operation was satisfactory.
April 13, 1986	PORV 455C cycled open due to an increase in reactor coolant system pressure during adjustment of reactor coolant system pressure. PORV actuation was satisfactory.
April 13, 1986	PORV 455C and 456 cycled open when the 4A charging pump was started. PORV operation was satisfactory.

APPENDIX B

ABSTRACT

An inservice Multi-Frequency Eddy Current Examination of Steam Generators # 4A, # 4B and # 4C at the Turkey Point Nuclear Plant was performed during the period of March 12 through March 16, 1986. The examination was conducted by the PNS-MCI Group and supplemented by Florida Power and 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 examination was recorded on magnetic tapes and also on two channel strip chart recorders. The recordings were evaluated by the data analyst 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 995 tubes (331 in # 4A, 318 in # 4B and 346 in # 4C) were inspected with no tube exhibiting an indication exceeding Plant Technical Specifications for plugging criteria.

Steam Generator # 4A:

Total tubes with indications $\geq$ 20% to 39% 0
Total tubes with indications $\geq$ 40% to 100% 0
Tubes plugged this outage 0
Tubes previously plugged 15
Total tubes plugged 15

Steam Generator # 4B:

Total tubes with indications $\geq$ 20% to 39% 1
Total tubes with indications $\geq$ 40% to 100% 0
Tubes plugged this outage 0
Tubes previously plugged 7
Total tubes plugged 7

Steam Generator # 4C:

Total tubes with indications $\geq$ 20% to 39% 2
Total tubes with indications $\geq$ 40% to 100% 0
Tubes plugged this outage 0
Tubes previously plugged 9
Total tubes plugged 9

Indication Locations:	S/G # 4A	S/G # 4B	S/G # 4C
AV Bars	0	0	0
Drilled Supports 1 - 6	0	1	2
Top of tube sheet to 1st support ..	0	0	0

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

PAGE 1 OF 1

EDDY CURRENT EXAMINATION RESULTS

PLANT: TURKEY POINT UNIT NO. 4

EXAMINATION DATES: MARCH 12, 1986 THRU MARCH 16, 1986

STEAM GENERATOR NUMBER	TOTAL TUBES INSPECTED	TOTAL INDICATIONS > OR = TO 20% TO 39%	TOTAL INDICATION > OR = TO 40% TO 100%	TOTAL TUBE PLUGGED AS PREVENTIVE MAINT	TOTAL TUBES PLUGGED
4A	331	0	0	0	0
4B	318	1	0	0	0
4C	346	2	0	0	0

CATION 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
4A	0	0	0	0	0
4B	0	1	0	0	0
4C	0	1	1	0	0

CERTIFICATION OF RECORD

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

FLORIDA POWER & LIGHT COMPANY
(Organization)

DATE 4/1/86 BY [signature]

PLANT CHANGE/MODIFICATION CPWO 85-164 PC/M CLASSIFICATION: NS

UNIT: 3

TURNED OVER DATE: 02-18-86

SUMMARY DATE: 06-16-86

REVISION: 0

ICW PUMP CHECK VALVE AIR CLOSING CYLINDER CLEVIS PINS

Summary:

This modification replaced the air closing cylinder clevis pins on the ICW pump check valve from cadmium plated carbon steel to 316 stainless steel for corrosion resistance.

Safety Evaluation:

This modification is nuclear safety related with no unreviewed safety question. The modification and analysis has ensured that the design criteria of the original system design document have been met and no accident or malfunction probability increased.

(ii) PROCEDURE CHANGES

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

a)	AP 0103.5:	Administrative Control of Valves, Locks and Switches
b)	AP 0103.19:	Monthly Verification of Safety Related System Flowpaths
c)	AP 0103.32:	Reactor Cold Shutdown Conditions
d)	ONOP 0208.12:	Annunciator List - Panel X - Common
e)	OP 2870.1:	Chemical and Volume Control System - Boric Acid
f)	OP 2879.1:	Waste Disposal System - Boric Acid Evaporator Package, Emergency Waste Processing
g)	OP 3204.1:	Residual Heat Removal System - Periodic Test
h)	ONOP 3208.1:	Malfunctional of Residual Heat Removal System
i)	OP 4004.1:	Containment Spray Pumps - Periodic Test
j)	OP 4104.2:	Engineered Safeguards and Emergency Power Systems - Integrated Test
k)	3(4)-OP-041.1:	Reactor Coolant Pump
l)	3(4)-OP-041.2:	Pressurizer Operation
m)	3(4)-OP-047:	CVCS - Charging and Letdown
n)	3(4)-OP-055:	Emergency Containment Cooling and Filtering Systems
o)	3(4)-OSP-030.2:	Operability of RCP Seal Cooling Water Outlet, FCV-3(4)-626
p)	3(4)-ONOP-030:	Loss of Component Cooling Water
q)	3-OP-030:	Component Cooling Water System

The procedures listed above were revised on March 28, 1986

r) 4-OP-030: Component Cooling Water System

Date of Change: April 10, 1986

Safety Evaluation Summary: 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.

2) The Emergency Operating Procedures (EOPs) were rewritten to incorporate the recommendations of the Westinghouse Emergency Response Guidelines (ERGs). The following procedures were written:

a) 3(4)-EOP-E-0: Reactor Trip or Safety Injection

b) 3(4)-EOP-E-1: Loss of Reactor Secondary Coolant

c) 3(4)-EOP-E-2: Faulted Steam Generator Isolation

d) 3(4)-EOP-E-3: Steam Generator Tube Rupture

e) 3(4)-EOP-ECA-0.0: Loss of all AC Power

f) 3(4)-EOP-ECA-0.1: Loss of all AC Power Recovery Without SI Required

g) 3(4)-EOP-EC-0.2: Loss of all AC Power Recovery With SI Required

h) 3(4)-EOP-ECA-1.1: Loss of Emergency Coolant Recirculation

i) 3(4)-EOP-ECA-1.2: LOCA Outside Containment

j) 3(4)-EOP-ECA-2.1: Uncontrolled Depressurization of All Steam Generators

k) 3(4)-EOP-ECA-3.1: SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

l) 3(4)-EOP-ECA-3.2: SGTR With Loss of Reactor Coolant - Saturated Recovery Desired

m) 3(4)-EOP-ECA-3.3: SGTR Without Pressurizer Pressure Control

n) 3(4)-EOP-ES-0.0: Rediagnosis

o) 3(4)-EOP-ES-0.1: Reactor Trip Response

p) 3(4)-EOP-ES-0.2: Natural Circulation Cooldown

q) 3(4)-EOP-ES-0.3: Natural Circulation Cooldown With Steam Void in Vessel (With RVLMS QSPDS)

r) 3(4)-EOP-ES-0.4: Natural Circulation Cooldown With Steam Void in Vessel (Without RVLMS)

s) 3(4)-EOP-ES-1.1: SI Termination

t) 3(4)-EOP-ES-1.2: Post LOCA Cooldown and Depressurization

u) 3(4)-EOP-ES-1.3: Transfer to Cold Leg Recirculation

v) 3(4)-EOP-ES-1.4: Transfer to Hot Leg Recirculation

w) 3(4)-EOP-ES-3.1: Post - SGTR Cooldown Using Backfill

x) 3(4)-EOP-ES-3.2: Post - SGTR Cooldown Using Blowdown

y) 3(4)-EOP-ES-3.3: Post - SGTR Cooldown Using Steam Dumps

z) 3(4)-EOP-F-0: Critical Safety Function Status Trees

aa) 3(4)-EOP-FR-C.1: Response to Inadequate Core Cooling

bb) 3(4)-EOP-FR-C.2: Response to Degraded Core Cooling

cc) 3(4)-EOP-FR-C.3: Response to Saturated Core Cooling

dd) 3(4)-EOP-FR-H.1: Response to Loss of Secondary Heat Sink

ee) 3(4)-EOP-FR-H.2: Response to Steam Generator Overpressure

ff) 3(4)-EOP-FR-H.3: Response to Steam Generator High Level

gg) 3(4)-EOP-FR-H.4: Response to Loss of Normal Steam Release Capabilities

hh) 3(4)-EOP-FR-H.5: Response to Steam Generator Low Level

ii) 3(4)-EOP-FR-I.1: Response to High Pressurizer Level

jj) 3(4)-EOP-FR-I.2: Response to Low Pressurizer Level

kk) 3(4)-EOP-FR-I.3: Response to Voids in Reactor Vessel

ll) 3(4)-EOP-FR-P.1: Response to Imminent Pressurized Thermal Shock Condition

mm) 3(4)-EOP-FR-P.2: Response to Anticipated Pressurized Thermal Shock Condition

nn) 3(4)-EOP-FR-S.1: Response to Nuclear Power Generation/ATWS

oo) 3(4)-EOP-FR-S.2: Response to Loss of Core Shutdown

pp) 3(4)-EOP-FR-Z.1: Response to High Containment Pressure qq) 3(4)-EOP-FR-Z.2: Response to Containment Flooding

rr) 3(4)-EOP-FR-Z.3: Response to High Containment Radiation Level

Safety Evaluation Summary: 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: March 31, 1986

3) An evaluation was performed to raise the setpoint of automatic closure of valves MOV-*-750 and MOV-*-751 to a pressure greater than any spurious spikes of the reactor coolant system/residual heat removal systems during solid water heatup of the reactor coolant system. The following procedures were changed:

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

   b) ONOP 0208.3: Annunciator List - Panel A - Reactor Coolant

   c) OP 3206.2: RHR System - Refueling Interval

   d) 3(4)-ONOP-050: Loss of RHR

   Safety Evaluation Summary: 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: December 5, 1985

4) IE Information Notice 85-94, "Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA", describes a recent instance in which the minimum flow requirements might not or could not be met for some emergency core cooling system pumps under small break loss-of-coolant-accident conditions. The events described by IEN 85-94 were determined to apply to Turkey Point Units 3 and 4 so the following procedures were revised to incorporate the temporary fix:

   a) AP 0103.19: Monthly Verification of Safety Related System Flowpaths

   b) OP 4004.1: Containment Spray Pumps - Periodic Test

c) OP 16002.6: Preparations and Precautions for Refueling Fuel Shuffle

d) EOP 20001: Loss of Reactor Coolant

Safety Evaluation Summary: 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: January 27, 1986

5) Due to the inoperability of containment isolation valves in the steam generator blowdown lines, an evaluation was performed to relocate the containment isolation boundary. The following procedures were revised:

a) AP 0103.5: Administrative Control of Valves, Locks, and Switches

b) 3(4)-OP-071: Steam Generator Blowdown Recovery System

Safety Evaluation Summary: 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: August 24, 1985

6) An evaluation determined that a failure of Intake Cooling Water Valve CV-*-2201 to close during an accident could result in less than design flow requirements to the component cooling water heat exchangers for an accident. Based on this the following procedure changes were made:

a) ONOP 3408.1: Intake Cooling Water Malfunction

b) EOP 20000: Immediate Actions and Diagnostics

c) EOP 20004: Loss of Offsite Power

Date of Changes: February 14, 1986

Safety Evaluation Summary: The proposed procedure changes require isolation of CV-*-2201 in the event of an accident with only one ICW pump operable. This will assure adequate ICW flow is directed to the CCW heat exchangers when only one ICW pump is available. 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.

7) The preliminary findings from an ongoing study of Emergency Diesel Generator loading indicated that the potential existed for overloading the emergency diesel generators when certain non-essential loads were automatically loaded onto vital busses following a safety injection actuation event coincident with a loss of offsite power. Based on this the following procedures were revised:

a) EOP 20000: Immediate Actions and Diagnostics

b) EOP 20004: Loss of Offsite Power

Date of Changes: December 15, 1985

c) EOP 20004: Loss of Offsite Power

Date of Change: January 2, 1986

Safety Evaluation Summary: The safety evaluation concluded that implementation of the administrative controls specified in the evaluation will assure safe operation of both Turkey Point Units 3 and 4. 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.

8) A revision to the evaluation described in Item 7 above resulted in revising the following procedures:

a) EOP 20000: Immediate Actions and Diagnostics

b) EOP 20004: Loss of Offsite Power

Safety Evaluation Summary: The safety evaluation concluded that implementation of the administrative controls specified in the evaluation will assure safe operation of both Turkey Point Units 3 and 4. 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 Change: December 18, 1985

9) As a result of an ongoing emergency diesel generator loading study additional administrative controls were identified that required revising the following procedures:

a) ONOP 9108.1: Main Transformer - Malfunction

b) 4-ONOP-004: Loss of Offsite Power

c) 3(4)-OP-007: 480 Volt Motor Control Centers

Safety Evaluation Summary: The evaluation ensures that sufficient emergency diesel generator capability exists to power the minimum safety equipment necessary for accident scenarios described in the FSAR including a full spectrum of breaks. 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: May 22, 1986

10) The following procedure was revised to correct a valve number. The valve number in FSAR Figure 6.2-1a does not agree with the number of the valve in the field:

a) 3(4)-OP-050: Residual Heat Removal System

Date of Change: October 25, 1985

Safety Evaluation Summary: The procedure change only involves correcting a valve number. 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.

11) Revision 1 to the evaluation for the failure of CV-*-2201 to close during an accident (see procedure change 6) provided additional guidance on the operation of CV-*-2201. The following procedure was revised:

a) OP 3400.1: Intake Cooling Water System - Normal Operation

Safety Evaluation Summary: The additional guidance will ensure adequate ICW flow is directed to the CCW heat exchangers when only one ICW pump is available. 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.

12) A design review of the proposed modifications to Unit 4 to the motor control center (MCC) 'D' automatic transfer scheme discovered a condition in which the transfer of MCC'D' from a potentially operable bus to a potentially inoperable bus could occur. The following procedures were revised as a result:

a) 3(4)-EOP-E-0: Reactor Trip or Safety Injection

b) ONOP 10308.1: Control Building Heating, Ventilation and Air Conditioning System

Date of Change: May 5, 1986

Safety Evaluation Summary: The implementation of the operator actions described in the safety evaluation will ensure that the FSAR design basis is met. 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.

13) The following procedure was revised to add more detail and clarify requirements:

a) NC-103: Diesel Fuel Oil Inventory Receiving of Shipment and Periodic Sampling

Date of Change: November 6, 1985

Safety Evaluation Summary: The testing that is done by the procedure does not include all of the testing stated in the FSAR referenced standards. The testing that is not done because of the difference in specifications, does not adversely affect safety. The tests that are not performed are not applicbale to the Turkey Point environment due to the relatively high year round temperatures. The testing that is completed provides adequate assurance of the fuel oil acceptance.

14) In order to incorporate the requirements of Technical Specification Amendment 11[illegible] (Facility Operating License DPR-1 for Unit 3) and Amendment 107 (Facility Operating License DPR-41 for Unit 4) which deleted the maximum amunt of enriched fissionable material which can be used in the core, or available on site, in the form of fabricated neutron flux detectors for the purpose of monitoring core neutron flux, the following procedure was changed:

a) OP 11550.43: Inventory and Leak Testing of Sealed Sources

Date of Change: July 3, 1985

15) The following procedure was revised to provide additional guidance for the new spent fuel pool storage racks:

    a) OP 16002.7: Refueling Shuffle in the Spent Fuel Pit

    Date of Change: October 3, 1985

    Safety Evaluation Summary: 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.

16) The following procedure was revised to provide guidance on the actions to be taken when process radiation monitor R-18 is inoperable:

    a) OP 5163.2: Waste Disposal System - Controlled Liquid Release to the Circulating Water

    Date of Change: October 8, 1985

    Safety Evaluation Summary: Technical Specification states that radioactive liquid releases may be permitted if R-18 is inoperable as long as certain sampling conditions are adhered to. Although, it is not explicitly stated in the FSAR that releases may occur if R-18 is inoperable, there is not a statement that prohibits it. The proposed change does not change the conclusion reached by FSAR Section 14.2.2 which is "No credible mechanism for accidental release of waste liquids to Biscayne Bay".

17) The following procedure was revised to include the proper temperature limits for the differential temperature between the reactor coolant system and pressurizer for both units:

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

    Date of Change: December 4, 1985

    Safety Evaluation Summary: The increase in differential temperature between the pressurizer and reactor coolant system to 320 degrees Fahrenheit for Unit 3 is acceptable from a technical and safety standpoint.

(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)
85-09	Pressurizer Heat Balance Test	3
85-10	CCW Heat Exchanger Performance Testing	3 and 4
85-13	Instrument Air System 4S Compressor Test	3 and 4
85-14	AFW N_2 Backup System Operability Verification; Unit 3 Train 1	3
85-16	AFW System - Special Governor Test	3 and 4
86-01	Component Cooling Water System Flow Test	3 and 4
86-02	"C" AFW Pump - Performance Test	3
86-03	Component Cooling Water System RHR Heat Exchanger Throttling Valve Adjustment Test	3 and 4
86-04	Intake Cooling Water System Flow Test	3
86-05	Component Cooling Water System RHR Heat Exchanger Throttling Valve Adjustment	3
86-08	Diesel Generator Meter Accuracy Testing	3 and 4
86-11	CCW System Flow Balancing	4
86-14	Air Conditioning Capabilities for the Control and Computer Rooms	3 and 4
86-15	Intake Cooling Water System Flow Test	4
86-16	Additional Emergency Diesel Generator Loading Data Collection	4
86-17	Intake Cooling Water System Flow Test	3

SPECIAL TEST 85-09

PRESSURIZER HEAT BALANCE TEST

Background Information:

This is a special test devised to determine the pressurizer mini-spray flow rate by performing a heat balance on the pressurizer.

Test Results:

Test was run January 5, 1986 on Unit 3, and is being evaluated by Engineering.

Safety Evaluation:

Pressurizer Heat Balance Special Test Safety Analysis.

This safety analysis addresses the temporary manual control of pressurizer heaters and spray valves to achieve near equilibrium conditions in the pressurizer to determine mini-spray flow rate from a heat balance calculation.

This test does not involve an unreviewed safety question because:

1. a) With respect to the probability of occurrence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident.

   b) With respect to the consequences of an accident previously evaluated in the FSAR: No accident would be made more serious during the performance of this test. The operator will be able to manually start equipment during the performance of this test should the need arise.

   c) With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of equipment malfunctioning will not be increased since the individual equipment will not be modified.

   d) With respect to the consequences of malfunction of equipment important to safety previously evaluated in the FSAR: The seriousness of equipment malfunction would not be increased as an operator will be able to start equipment should the need arise.

2. a) With respect to the possibility of an accident of a different type than any analyzed in the FSAR: This test does not create the possibility of an accident that is not considered in the FSAR because all of the equipment affected is considered in the FSAR.

   b) With respect to the possibility of malfunction of a different type than any analyzed in the FSAR: All of the equipment involved in the test or affected by the test is considered in the FSAR. The test does not create the possibility of malfunction of any equipment other than those considered in the FSAR.

3. With respect to the margin of safety as defined in the basis for any Technical Specification: The margin of safety will not be decreased, since an operator will be able to restore the equipment to its normal lineup if required.

SPECIAL TEST 85-10

CCW HEAT EXCHANGER PERFORMANCE TESTING

Background information:
The purpose of this test was to collect fouling rate data that would be used to determine the frequency at which the CCW heat exchanger must be cleaned. This data would also provide a measure of the effectiveness of the cleaning process. In addition, the results of this test provides the maximum allowable ICW inlet temperature, and the number of heat exchangers that are required for a given value of ICW inlet temperature. This test will become a periodic test to monitor fouling of these heat exchangers.

Test Results:
Special Test 86-10 was performed on the Units 3 and 4 CCW heat exchanges on the following dates:

UNIT NUMBER	TEST DATE
3	8-15-85
4	8-15-85
3	9-05-85
4	9-05-85
3	11-20-85
4	11-20-85
3	3-03-86
3	3-12-86
3	3-19-86
3	3-27-86
3	4-02-86
3	4-10-86
3	4-25-86
3	5-01-86
3	6-19-86
3	7-29-86

The early tests indicated excessive fouling in the heat exchangers. They were then cleaned and the tests were repeated to evaluate the cleaning process. Using this data, criteria were developed for cleaning the heat exchanges based on the number of days since the last cleaning, the number of heat exchangers in operation, and the maximum ICW inlet temperature. Additionaly, R-Value vs days since last cleaning curves, and Heat Transfer Coefficient vs days since last cleaning curves for 3 flow rates were developed for both units. The results of the periodic repetition in of this test will be used to develop long term solutions to the CCW Heat Exchanger fouling problems.

Safety Evaluation:
This test does not involve and unreviewed safety question because:

1. a. With respect to the probability of occurence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: No accident would be made more serious because the system is designed with 3-50% Heat Exchangers and 3-100% pumps. Only one Heat Exchanger is being valved out of service at a time and an operator will be on hand to place the equipment back in service should the need arise.

c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since the individual equipment will not be modified.

d. With respect to the consequence of malfunction of equiment important to Safety previously evaluated in the FSAR: The seriousness of equipment malfunction would not be increased as an operator will be assigned to the CCW area to place the equipment in normal operating condition should it be required.

2. a. With respect to the possibility of an accident of a different type than any analized in the FSAR: This test does not create the possibility of an accident that is not considered in the FSAR because all the equipment affected is considered in the FSAR.

b. With respect to the possibility of malfunction of a different type than any analyzed in the FSAR: All of the equipment involved in the test or affected by the test is considered in the FSAR.
The test does not create the possibility of malfunction of any equipment other than those considered in the FSAR.

c. With respect to the margin of safety as defined in the basis for any Technical Specification: The margin of safety will not be decreased because the Technical Specification permits the removal of one Heat Exchanger for limited periods of time while the unit is operating. An operator will be in the area to place the equipment into operation if necessary.

SPECIAL TEST 85-13

UNITS 3&4 INSTRUMENT AIR SYSTEM 4S COMPRESSOR TEST

Background Information:
This is a Special Test devised to evaluate the performance of the 4S Instrument Air Compressor, and to collect data, which would help determine changes to be made to the compressor in order to improve its performance.

Test Results:
The test was run 10-10-85. The Tech. Dept used the data to improve compressor performance.

Safety Evaluation:
Special Test of the 4S Instrument Air Compressor.

This safety analyses addressed the impact performing the Special Test on the Instrument Air System.

The Instrument Air System is not a nuclear safety related system. The 4S compressor had been idled for several months. Prior to the test, the Instrument Air System was operating satisfactory as per original plant design of 3 stationary electric driven compressors and one backup emergency diesel compressor. The test obtained motor amps, discharge pressure, header pressure, at no load, half load and full load compressor operation. Analyses of that data resulted in recommendations that returned the compressor to service, and improved Instrument Air System reliability.

Based on the above it can be said that Special Test 85-13 did not involve nor will it involve an unreviewed safety question.

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SPECIAL TEST 85-14

AUXILIARY FEEDWATER N_2 BACKUP SYSTEM OPERABILITY VERIFICATION; UNIT 3 TRAIN 1

Performed: November 15, 1985

BACKGROUND INFORMATION:

This test was performed to verify the N_2 Low Pressure Alarm Setpoint of 1350 psig gives an operator adequate time to valve in an additional N_2 bottle; for the worst case, flow control valves in AUTO mode, for Unit 3 Train 1.

TEST RESULTS:

This testing concluded that the N_2 backup low pressure alarm setpoint of 1350 psig provides adequate time for an operator to valve in an additional N_2 bottle following a N_2 low pressure annunciator alarm.

SAFETY EVALUATION:

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 or any AFW equipment important to the function or the 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 85-16

AUXILIARY FEEDWATER SYSTEM - SPECIAL GOVERNOR TEST

Performed: January - February 1986

BACKGROUND INFORMATION:

This special test was conducted to study the performance of the Auxiliary Feedwater Turbine Governor and the turbine overspeed protection devices.

TEST RESULTS:

Testing was completed in February 1986 and the data transmitted to Juno Beach Engineering for evaluation per letter PTN-TECH-86-496.

SAFETY EVALUATION:

During portions of this special test, the electronic overspeed trip will be disconnected. The mechanical overspeed trip remains connected throughout the duration of this test and provides protection against turbine overspeeds, which could jeopardize equipment integrity or personnel safety. To provide a second means of protection, the test operators are instructed not to allow pump/turbine speed to exceed 6600 RPM when the electronic overspeed trip is disconnected. During the controlled overspeed portion of this test, the operator will use the overspeed device to slowly allow the turbine speed to approach the mechanical overspeed trip setpoint.

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. When performed during a single plant outage, this special AFW Governor Test does not reduce the margin of safety discussed in the Technical Specifications.

SPECIAL TEST 86-01

COMPONENT COOLING WATER SYSTEM FLOW TEST

Background Information

The purpose of the CCW System Flow Test was to provide assurance that the design CCW flow rates can be achieved under post-LOCA recirculation phase conditions assuming loss of one electrical train. The test is configured to verify engineering flow calculations under various CCW and RHR system alignments.

In order to demonstrate system flow capacity, the non-operating unit was configured with CCW flow through one RHR heat exchanger, one of three CCW pumps running, three CCW heat exchangers aligned and both CCW headers in operation. The test was performed using several combinations of system alignments of Trains A and B to verify that worse case conditions would be evaluated.

Test Results

Unit 4 was modified per OTSC #3987 on March 3, 1986 to perform this test. Results indicated that a full open position on RHR heat exchanger 4-A outlet valve 4-748A provided excessive flow through the RHR heat exchanger. This resulted in insufficient flow through the Emergency Containment Coolers under the system alignment conditions during the test. Subsequently, Special Test 86-03 was written to determine proper RHR throttle valve position.

Safety Evaluation

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

1. a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident. This test will be performed while unit 4 is in a refueling outage; some fuel may be in the reactor vessel since the test will be performed during the fuel load cycle. This test merely simulates the CCW system configuration during the recirculation phase of a LOCA and is, therefore, only providing confirmation that the system meets the FSAR requirements.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: The only safety-related loads serviced by the CCW system during the test are the Spent Fuel Pit (SFP) heat exchanger, RHR heat exchangers and RHR pump seal water heat exchangers. One step of the test requires short term isolation of the cooling water flow to the SFP heat exchanger. Conservative extrapolations of engineering calculations, considering the recent Unit 4 full core offload, indicate that SFP temperature rise will be less than 9°F/HR, with the heat exchanger isolated. SFP temperature will be monitored during the test. If the SFP temperature should increase beyond 170°F, an operator will have sufficient time to return the SFP heat exchanger to service and maintain the SFP temperature within the maximum normal operating temperature of 183°F. Also, a calculation has been performed to determine the effect of stopping cooling to the RHR heat exchanger with the core reloaded and the heat up rate of the reactor cavity is approximately 6°F/HR. Therefore, the Technical Specification limit specified in Table 1.1 for refueling (Operational Mode 6) of 140°F will not be exceeded by any short term interruption of cooling during the test.

At the time of this test, the containment will be vented to the atmosphere. It is not considered credible to postulate containment temperature rising to the normal operating design temperature of 120°F.

One CCW pump will be running continuously during the test. There is no credible situation which would require the immediate start of a second CCW pump. However, a second CCW Pump will be available for continued system operation.

c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since the individual equipment will not be modified. During the test, the CCW system will be aligned for full CCW flow through one RHR heat exchanger, the possibility exists that the CCW pump may reach flow rates which exceed the maximum flow rates documented on the certified pump curves. However, Net Positive Suction Head (NPSH) calculations have been performed to demonstrate that adequate NPSH will be available under all test conditions and formal correspondence is available from the pump manufacturer (Worthington Corporation) which states that provided adequate NPSH is available the pump may be run at runout conditions without damage. Futhermore, if the CCW pump being tested should fail a second CCW pump could be started from the control room to maintain flow. As described in 1.b above, the safety-related loads which requires cooling during the test are the spent fuel pit heat exchanger, RHR heat exchanger and RHR pump seal water heat exchanger. Should the CCW pump being tested fail, there is adequate time to re-establish CCW flow before the spent fuel pit water reaches the maximum normal operating temperature of 183°F or the reactor cavtiy or core outlet temperature reach their Technical Specification limit.

d. With respect to the consequences of malfunction of the equipment important to safety previously evaluated in the FSAR: During the test, the CCW system will be aligned and operated within its design basis limitations. Therfore, the consquences of malfunction of the equipment important to safety will be no different from that previously evaluated in the FSAR. As described in 1.c above, one CCW pump may exceed the maximum flow rate documented on the certified pump curves. In the unlikely event of a CCW pump failure, at least one of the 2-100 percent capcity CCW pumps will be available for continued system operation.

2. a. With respect to the possibility of an accident of a different type than any analyzed in the FSAR: This test simulates the configuration and operability of the CCW System which would exist during the recirculation phase following LOCA conditions so that data can be obtained for the purpose of evaluating system performance, particularly CCW flow through the RHR heat exchanger. Data obtained from this test will also be used to determine if current system alignment and performance is acceptable with respect to FSAR design basis requirements.

The test will be performed during a Unit 4 Refueling Outage. Short term isolation of CCW flow to the SFP heat exchanger is discussed in 1.b above. Also, in the unlikely event of a CCW pump failure, there is sufficient time to place another CCW pump in operation to provide flow to the SFP heat exchanger, RHR heat exchanger and RHR pump seal water heat exchangers.

Based upon the preceeding, the possibility of an accident of an a different type than any analyzed in the FSAR will not be created.

b. With respect to the possibility of malfunction of a different type than any analyzed in the FSAR: The proposed test simulates CCW system alignment and performance which would exist during post LOCA recirculation, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication with the control room operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

3. With respect to the margin of safety as defined in the basis for any Technical Specification: The margin of safety will not be decreased because adequate precautions are noted in the test procedure which will assure that all applicable Technical Specification limits will be maintained during the performance of this test.

2:016

SPECIAL TEST 86-01, CHANGE REQUEST NO. 1

AS FOUND CCW FLOW TEST UNIT 3

Background Information:
Change request No.1 to Special Test 86-01 was written to obtain as found data on Unit-3 CCW system. Special Test 86-01 was originally written for testing of Unit-4 in refueling shutdown (Mode 6) conditions. CR No. 1 provided for testing of Unit-3 under cold shutdown (Mode 5) conditions.

Test Results
Special Test 86-01, CR No.1 was performed on Unit 3 on March 5, 1986. Results of the test were essentially identical to the results seen when testing Unit 4.

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

1. a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident. This test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). This test simulates the CCW system configuration during the recirculation phase of a LOCA and is, therfore, only providing confirmation that the system meets the FSAR flow requirements.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: The safety-related loads serviced by the CCW system while in Mode 5 are the Spent Fuel Pit (SFP) heat exchanger, RHR heat exchangers and RHR pump seal water heat exchangers. Component cooling water flow to one RHR heat exchanger will be maintained throughout the duration of the test.

   At the time of this test, the containment may be vented to the atmosphere since the normal containment coolers will be isolated. It is not considered credible to postulate containment temperature rising to the normal operating design temperature of 120°F since the reactor coolant pumps are not operating and the plant is in a cold shutdown condition.

   One CCW pump will be running continously during the test. There is no credible situation which would require the immediate start of a second CCW pump. However, either "standby" CCW pump will be available for continued system operation if required.

   c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since the individual equipment will not be modified. During the test, the CCW system will be aligned and operated within its design basis configurations. Furthermore, if the CCW pump being tested should fail, a second CCW pump could be started from the control room to maintain flow . As described in 1.b above, the safety-related loads which require cooling during the test are the spent fuel pit heat exchanger, RHR heat eschanger and RHR pump seal water heat exchangers. Should the CCW pump being tested fail, there is adequate time to reestablish CCW flow before the spent fuel pit water reaches the maximum normal operating temperature of 183°F, exceed the seal water temperature limit of the RHR pumps or exceed the Technical Specification RCS temperature limit for Mode 5.

d. With respect to the consequences of malfunction of the equipment important safety previously evaluated in the FSAR: During the test, the CCW system will be aligned and operated within its design basis limitations. Therefore, consquences of malfunction of the equipment important to safety will be no different from that previously evaluated in the FSAR. In the unlikely event of a CCW pump failure, two additional 100 percent capacity CCW pumps will be available for continued system operation.

2. a. With respect to the possibility of an accident of a different type than any analyzed in the FSAR: This test simulates the configuration and operability of the CCW System which would exist during the recirculation phase following LOCA conditions so that data can be obtained for the purpose of evaluating system performance.

   The test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). In the unlikely event of a CCW pump failure, there is sufficient time to place another CCW pump in operation to provide flow to the SFP heat exchanger, RHR heat exchanger and RHR pump seal water heat exchangers.

   Based upon the preceding, the possibility of an accident of a different type than any analyzed in the FSAR will not be created.

   b. With respect to the possibility of malfuction of a different type than any analyzed in the FSAR: The proposed test simulates CCW system alignment and performance which would exist during post LOCA recirculation, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication with the control room operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

3. With respect to the margin of safety as defined in the basis for any Technical Specification: The margin of safety will not be decreased because adequate precautions are noted in the test procedure which will assure that all applicable Technical Sepcifications limits will be maintained during the performance of this test.

SPECIAL TEST 86-02

"C" AUXILIARY FEEDWATER PUMP - PERFORMANCE TEST

Performed: May 6, 1986

BACKGROUND INFORMATION:

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

TEST RESULTS:

The study concluded that the underfiled impeller replacement improved the pump performance to above the design requirements.

SAFETY EVALUATION:

In Section 9.7 of this Special Test, the "C" 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 20-377 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 governor 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 the turbine lube oil cooler 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 86-03
COMPONENT COOLING WATER SYSTEM RHR HEAT EXCHANGER THROTTLING VALVE ADJUSTMENT TEST

Background Information:
In Special Test 86-01 the flow balance of the CCW system with RHR heat exchanger throttle valves wide open was found to be unsatisfactory. Special Test 86-03 was written and performed to demonstrate the ability to balance flow in the CCW System.

In this test, a series of valve manipulations and flow test configurations were performed to establish acceptable RHR heat exchanger throttle valve positions. At least 4250 GPM CCW flow through one RHR heat exchanger is required with one CCW pump, two CCW heat exchangers, and both CCW heat exchangers operating. This will adequately simulate post-LOCA recirculation phase flow conditions.

The following on test to this is Special Test 86-05 which then readjusts the CCW System Flow Balance on unit 3 based on the data from this test. Similar flow balancing was done on Unit 4 via Special Test 86-11.

Test Results
Special Test 86-03 was begun on March 7, 1986, however the test was canceled after doing "A" Train flow balance only. Special Test 86-03 was modified by OTSC 3997 to leave the Boric Acid Evaporator valved in. This test was performed on a train only on March 8, 1986.

After evaluation of the data from train "A", the test was performed on the "B" train portion of the system on March 9, 1986. Results of the test demonstrated adequate balancing of CCW flow with one or two CCW pumps operating in the post-LOCA recirculation mode.

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

1. a. With respect to the probability of occurence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident. This test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). This test simulates the CCW system configuration during the recirculation phase of a LOCA and, therefore, will provide confirmation that the system meets the FSAR flow requirements.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: The safety-related loads which require flow from the CCW system while in Mode 5 are the Spent Fuel Pit (SFP) heat exchanger, RHR heat exchangers and RHR pump seal water heat exchangers. The test requires short term isolation of the cooling water flow to the SFP heat exchanger. Conservative engineering calculations, based on a one-half core offload with 37 days days decay and the remainder of the 1404 fuel storage spaces filled with fuel from prior fuel cycles, indicate that SFP temperature rise will be approximately 6°F/HR, with the heat exchanger isolated. SFP temperature will be monitored during the test. If the SFP temperature should increase beyond 170°F, an operator will have sufficient time to return the SFP temperature below the maximum normal operating temperature of 183°F.

   One RHR coolant loop will be operating and the other RHR coolant loop will be maintained operable throughout the duration of the test.

At the time of this test, the containment may be vented to the atmosphere since the normal containment coolers will be isolated. It is not considered credible to postulate containment temperature rising to the normal operating design temperature of 120°F since the Reactor Cooling pumps will be secured and the plant will be in a cold shutdown condition.

One CCW pump will be running continuously during the test. There is no credible situatuion which would require the immediate start of a second CCW pump. However, at least one standby CCW pump will be available to support continued system operation, if required.

c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since no equipment will be modified. However, this test will require adjustment of the in-service RHR heat exchanger throttling valve from its existing position to a new position which provides the design CCW flow. During the test, the CCW system will be aligned and operated within its design basis configuration. With one CCW pump operating and the system aligned for CCW flow through one RHR heat exchanger, the possibility exists that the CCW pump may reach flow rates which exceed the maximum flow rates documented on the certified pump curves. However, Net Positive Suction Head (NPSH) calculations performed demonstrate that adequate NPSH will be available under all test conditions and formal correspondence is available from the pump manufacturer (Worthington Corporation) which states that provided adequate NPSH is available, the pump may be run at runout conditions without damage. Furthermore, if the CCW pump being tested should fail, a second CCW pump could be started from the control room to maintain flow. As described in 1.b above, the safety-related loads which require cooling during the test are the spent fuel pit exchanger, RHR heat exchanger and RHR pump seal water heat exchangers. Should the CCW pump being used in the test fail, there is adequate time to re-establish CCW flow to maintain the normal operating temperature of the spent fuel pit, the RHR pump seal water within its temperature limit and the RCS temperature within the Mode 5 Technical Specification limit. Additionally, during the valve throttling process, the RCS temperature will be maintained below the Mode 5 Technical Specifications limit.

d. With respect to the consequences of malfunction of the equipment important to safety previously evaluated in the FSAR: During the the test, the CCW system will be aligned and operated within its design basis limitations. Therfore, the consequences of malfunction of the equipment important to safety will be no different from that previously evaluated in the FSAR. As described in 1.c above, one (1) CCW pump may exceed the maximum flow rate documented on the certified pump curves. In the unlikely event of a CCW pump failure , at least one (1) of the 2-100 percent capcity standby CCW pumps will be available to maintain system operation.

2. a. With respect to the possibility of an accident of a different type than any analyzed in the FSAR: This test simulates the configuration and operability of the CCW of the CCW system which would exist during the recirculation phase following a LOCA so that performance and for establishing the optimum postition for the RHR heat exchanger throttling valve(s).

The test will be performed while Unit 3 is in a cold shutdown condition (mode 5). Short term isloation of CCW flow to the SFP heat exchanger is discussed in 1.b above. Also, in the unlikely event of a CCW pump failure, there is sufficient time to place another CCW pump in operation to provide flow to the SFP heat exchangers, RHR heat exchanger and RHR pump seal water heat exchangers.

Based upon the proceding, the possibility of an accident or a different type than any analyzed in the FSAR will not be created.

b. With respect to the possibility of a malfunction of a different type than any analyzed in the FSAR: The proposed test simulates CCW system alignment and performance which would exist during post LOCA recirculation, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication with the control room operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

3. With respect to the margin of safety as defined in the basis for any Technical Specifications: The margin of safety will not be decreased because the test will be performed during a cold shutdown condition. In addition, adequate precautions are noted in the test procedure which will assure that all applicable Technical Specifications limits will be maintained during the performance of this test

SPECIAL TEST 86-04

INTAKE COOLING WATER SYSTEM FLOW TEST - UNIT 3

Background Information:

This test was performed to measure total Intake Cooling Water flow from one Intake Cooling Water Pump to all three Component Cooling Water Heat Exchangers and both Turbine Plant Cooling Water Heat Exchangers with the Isolation Valve to the Turbine Plant Cooling Water Heat Exchangers fully open.

Test Results:

This test was performed on March 9, 1986, and the test results were forwarded to Engineering for evaluation and review.

Safety Evaluation:

The performance of this test does not involve an unreviewed safety question nor does it increase the probability of an accident because:

The performance of this test does not require any system changes which would increase the probability of occurrence of an accident. This test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). This test simulates the Intake Cooling Water System configuration following a LOCA with a postulated failure to isolate the Turbine Plant Cooling Water (TPCW) Heat Exchangers. The purpose of this test is to verify that any one (1) Intake Cooling Water (ICW) pump will provide the post-LOCA required flow through the Component Cooling Water (CCW) Heat Exchangers, assuming flow through the Turbine Plant Cooling Water Heat Exchangers is not isolated.

The consequences of any accident previously evaluated in the FSAR is expected to be less severe, since this test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). In addition, as an additional conservatism, the CCW and ICW systems will be maintained operable in accordance with Technical Specifications 3.4-4 and 3.4-5 for power operation, throughout the course of the test.

The probability of a malfunction of equipment important to safety will not be increased based on incorporation of the following precautions into the test procedure:

1. Minimum required pump submergence will be ensured by verification of Intake Canal level before any ICW pump is started.

2. Lube water flow to each ICW will be maintained within specified limits.

3. ICW pump motor current will be monitored continuously from the Main Control Room; if normal ICW pump motor current is exceeded, which could indicate pump runout, a second ICW pump will be started.

4. In the event of excessive vibration, unusual noise, overheating or other abnormal symptoms, the test will be discontinued and corrective action taken.

The precautions listed above are intended to prevent pump damage due to pump runout, loss of lube water or operation with MPSH available less than MPSH required.

In addition, flow rates through the CCW and TPCW heat exchangers will be maintained below the maximum specified limits to prevent vibration induced tube damage.

Since each ICW pump will be tested separately with the two (2) remaining pumps secured, in the unlikely event of an ICW pump failure, the test will be terminated and the standby pumps started to maintain system operability.

During the test, the ICW system will be aligned and operated within its design basis limitations, with the exception of single ICW pump operation with the TPCW heat exchangers not isolated. As stated above, adequate precautions have been incorporated into the procedure to preclude pump damage. Since each ICW pump will be tested separately with the two (2) remaining pumps secured, in the unlikely event of an ICW pump failure, the test will be terminated and the standby pumps started to maintain system operability.

This test simulates the Intake Cooling Water System configuration following a LOCA with a postulated failure to isolate the Turbine Plant Cooling Water heat exchangers. Data obtained from this test will be used to verify that any one (1) ICW pump will provide the post-LOCA required flow through the CCW heat exchangers assuming flow through the TPCW heat exchangers is not isolated.

The test will be accomplished in a controlled manner in accordance with an approved procedure under the direct supervision of a dedicated Test Coordinator who will be in communication with the Control Room Operators. As such, system and equipment parameters will be monitored closely, with appropriate actions taken to prevent component damage and return the system to normal should operating limits be approached.

Based on the preceeding, the possibility of an accident of a different type than any analyzed in the FSAR will not be created.

The proposed test simulates ICW system alignment which could exist post LOCA, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication witht the Control Room Operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

The margin of safety will not be decreased because the CCW and ICW systems will be maintained operable in accordance with Technical Specifications 3.4-4 and 3.4-5 throughout the course of the test.

SPECIAL TEST 86-05

COMPONENT COOLING WATER SYSTEM RHR HEAT EXCHANGER THROTTLING VALVE ADJUSTMENT TEST

Background Information:

The purpose of this is to balance the CCW system to ensure that the essential components receive their minimum required flow rate during a MHA with the most limiting single failure assumed to occur.

Test Results:

The test was run on March 19, 1986, the results were evaluated by Engineering and the data from the test established the required position for critical throttle valves in the CCW system.

Safety Evaluation:

This safety evaluation addresses the temporary flow test configuration and associated valve manipulations of the CCW system.

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

1. a. With respect to the probability of occurrent of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident. This test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). This test simulates the CCW system configuration during the recirculation phase of a LOCA and, therefore, will provide confirmation that the system meets the FSAR flow requirements.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: The safety-related loads which require flow from the CCW system while in Mode 5 are the Spent Fuel Pit (SFP) heat exchangers, RHR heat exchangers and RHR pump seal water heat exchangers.

   One RHR coolant loop will be operating and the other RHR coolant loop will be maintained operable throughout the duration of the test.

   At the time of this test, the containment may be vented to the atmosphere since the normal containment coolers will be isolated. It is not considered credible to postulate containment temperature rising to the normal operating design temperature of 120°F since the Reactor Coolant pumps will be secured and the plant will be in a cold shutdown condition.

   One CCW pump will be running continuously during the test. There is not credible situation which would require the immediate start of a second CCW pump. However, at least one standby CCW pump will be available to support continued system operation, if required.

c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since no equipment will be modified. However, this test will require adjustment of the in-service RHR heat exchanger throttling valve from its existing position to a new position which provides the design CCW flow. During the test, the CCW system will be aligned and operated within its design basis configuration. With one CCW pump operating and the system aligned for CCW flow through one RHR heat exchanger, the possibility exists that the CCW pump may reach flow rates which exceed the maximum flow rates documented on the certified pump curves. However, Net Positive Suction Head (NPSH) calculations performed demonstrate that adequate NPSH will be available under all test conditions and formal correspondence is available from the pump manufacturer (Worthington Corporation) which states that provided adequate NPSH is available, the pump may be run at runout conditions without damage. Furthermore, if the CCW pump being tested should fail, a second CCW pump could be started from the control room to maintain flow. As described in 1.b above, the safety-related loads which require cooling during the test are the spent fuel pit heat exchanger, RHR heat exchanger and RHR pump seal water heat exchangers. Should the CCW pump being used in the test fail, there is adequate time to re-establish CCW flow to maintain the normal operating temperature of the spent fuel pit, the RHR pump seal water within its temperature limit and the RCS temperature within the Mode 5 Technical Specifications limit. Additionally, during the valve throttling process, the RCS temperature will be maintained below the Mode 5 Technical Specifications limit.

d. With respect to the consequences of malfunction of the equipment important to safety previously evaluated in the FSAR: During the test, the CCW system will be aligned and operated within its design basis limitations. Therefore, the consequneces of malfunction of the equipment important to safety will be no different from that previously evaluated in the FSAR. As described in 1.c above, one (1) CCW pump may exceed the maximum flow rate documented on the certified pump curves. In the unlikely event of a CCW pump failure, at least one (1) of the two (2) - 100 percent capacity standby CCW pumps will be available to maintain system operation.

2. a. With respect to the possibility of an accident of a different type than any analyzed in the FSAR: This test simulates the configuration and operability of the CCW system which would exist during the recirculation phase following a LOCA so that data can be obtained for the purpose of evaluating system performance and for establishing the optimum position for the RHR heat exchanger throttling valve(s).

The test will be performed while Unit 3 is in a cold shutdown condition (Mode 5). In the unlikely event of a CCW pump failure, there is sufficient time to place another CCW pump in operation to provide flow to the SFP heat exchanger, RHR heat exchanger and RHR pump seal water heat exchangers.

Based upon the preceeding, the possibility of an accident of a different type than any analyzed in the FSAR will not be created.

b. With respect to the possibility of a malfunction of a different type than any analyzed in the FSAR: The proposed test simulates CCW system alignment and performance which would exist during post LOCA recirculation, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication with the control room operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

3. With respect to the margin of safety as defined in the basis for any Technical Specifications: The margin of safety will not be decreased because the test will be performed during a cold shutdown condition. In addition, adequate precautions are noted in the test procedure which will assure that all applicable Technical Specfifications limits will be maintained during the performance of this test.

SPECIAL TEST 86-08

DIESEL GENERATOR METER ACCURACY TESTING

REFERENCE: DIESEL GENERATOR METER ACCURACY EVALUATION
PTPO-86-275-E-2

Background Information:

The purpose of this test was to determine the effects of metering errors on the EDG instrumentation provided at Turkey Point Units 3 and 4. Specifically, it was determined what effect these errors have on the capability of the EDG to supply the required safety loads without exceeding the capabilities of both the diesel engine and generator.

A test was performed on the wattmeter and ammeter instrumentation loops for each EDG. This test was performed by Multi-Amp Services with assistance by Ebasco Services and Florida Power & Light.

The test consisted of removing the current and potential transformers from the loops, impressing currents and voltages into the loop to simulate running the EDG's. High accuracy equipment was used during the test. The accuracy of the existing metering was then checked against the reference test equipment by taking multiple readings. In addition, tests were performed to determine that the current and potential transformers were not overburdened, allowing the use of the manufacturer's maximum error limitations.

Test Results:

The error of each loop was calculated using the published accuracy values for the test instrumentation, current transformers and potential transformers. This error has been determined to be approximately ±20KW for the wattmeter loops and ±3A for the ammeter loops. Also, an additional 20KW and 5A is considered to account for instrument drift.

It should be noted that the ±20KW and ±5A drift number is defined as that value of variance on the meter readings which is to be expected during the calibration intervals. The components within the metering loops have been applied within their design limitations and are, therefore, not expected to have appreciable drift. The variance is, however, very much a function of the ability of a person to resolve correctly the true meter reading. It is expected that a reading can be taken accurately to a value of ±10KW and ±1A.

Based on the above, it was recommended that the plant perform a loop calibration check for these meters on a monthly basis for a minimum of three months. This calibration interval may be extended based on favorable results of the calibrations.

The maximum loop tolerance has been determined to be approximately ±8A (3A error plus 5A drift). Therefore, the allowed loading should not exceed 470A which equates to 2879KW @.85pf.

Safety Evaluation:

This safety evaluation addresses the temporary test configuration to perform a meter accuracy test. The temporary configuration for this test does not involve an unreviewed safety question nor does it increase the probability of an accident because:

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

   No changes were made that would increase the likelihood of an accident. This test was performed while Units 3 and 4 were in a cold shutdown condition (Mode 5). This test simulates the configuration of the system during normal/accident conditions to provide confirmation that the system operates as required to allow safe shutdown.

   B. With respect to the consequences or probability of malfunction of equipment important to safety previously evaluated in the FSAR:

   The probability of malfunction of equipment was not increased since no equipment will be modified. This test, however, required that currents and voltages be injected into the components within the metering loop to verify their response to the test signals. These current and voltage signals were kept below published limitations and ratings of the affected equipment.

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

   The test was performed while Units 3 and 4 were in a cold shutdown condition (Mode 5). This test simulates the configuration of the system during normal accident conditions to provide confirmation that the system operates as required to allow safe shutdown.

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

   The margin of safety was not decreased because the test was performed in a cold shutdown condition. In addition, all Technical Specification limits were maintained during performance of this test.

Therefore, the above test does not involve an unreviewed safety question nor does it decrease the safety of the plant.

SPECIAL TEST 86-11

4 UNIT-4 CCW SYSTEM FLOW BALANCING

Background Information:
Special test 86-11 was the Unit-4 version of Special Test 86-05, performed on Unit-3. In addition, the test included additional KW loading data for studies of diesel loading. For this test, the RHR heat exchanger CCW flow isolation valve (MOV-4-749A) was out of service. Therefore, the throttle valve (MOV-4-748A) was used as the isolation valve, and the isolation valve was used to determine proper CCW flows needed to balance the systems.

Test Results:
Special tests 86-11 was completed on May 2, 1986, demonstrating proper CCW flows to the RHR heat exchangers. Upon completion of the test, the throttle valve (MOV-4-748A) was placed in its balanced position and tagged so that it would not be changed.

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

1. a. With respect to the probability of occurance of an accident previously evaluated in the FSAR: No change has been made to equipment in the component cooling water system. Therefore, the probability of an accident has not been increased.

   b. With respect to the consequences of an accident perviously evaluated in the FSAR: The consequences of an accident have not been increased. In the event that the 4A RHR heat exchanger is required, the dedicated operator stationed at 4-748A will have sufficient time to open 4-748A to its 30% throttled position (Approx. 22 turns).

   c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment important to safetv has not been increased. MOV-4-749A will be open with its power supply racked out. Valve 4-748A will be manned by a dedicated operator in constant contact with the control room. In the event that the 4A RHR heat exchanger is required, sufficient time exist to return 4-748A to its 30% throttled position.

   d. With respect to the consequences of malfunction of the equipment important to safety previously evaluated on the FSAR: The consequences of a malfunction of equipment have not been increased. MOV-4-749A will be open with its power supply racked out. Valve 4-748A will be manned by a dedicated operator in constant contact with the control room. In the event that the 4A RHR heat exchanger is required, sufficient time exist to return 4-748A to its 30% throttled position.

2. a. With respect to the possibility of an accident of a different type than previously analyzed in the FSAR: No change has been made to the component cooling water system; therefore, the possibility of an accident not analyzed has not been increased.

   b. With respect to the possibility of a malfunction of a different type than previously analyzed in the FSAR: No change has been made to the component cooling water system; therefore, the possibility of malfunction has not been increased.

3. a. With respect to the margin of safety as defined in the basis of Technical Specifications: The margin of safety has not been decreased because with a dedicated operator stationed at 4-748A and MOV-4-749A open with its power supply racked out, at no time will less than two reactor coolant loops be operable.

SPECIAL TEST 86-14

AIR CONDITIONING CAPABILITIES FOR THE CONTROL AND COMPUTER ROOMS

Background Information:
This is a Special Test designed by JPE to (a) evaluate the heat rise of the Computer and Cable Spreading Room after the air conditioning stops operating. (b) Evaluate whether the Control Room can operate with only one air condtioning unit.

These tests were made to help define Emergency Diesel Generator loading after a loss of offsite power accident occurs.

Test Results:
The test was performed, and the results forwarded to Juno for evaluation. Preliminary review showed that the tests were successful.

Safety Evaluation:
The test was performed with adequate safeguards so that ambient termperature limitations on the areas tested, were not exceeded. Care was taken at the Control Room to quickly determine the nature of any alarms of faulty equipment read out due to overheating of the instruments/equipment in the Control Room.

Based on the above, it can be said that Special Test 86-14 did not pose nor involve an unreviewed safety question.

LAM/lam/d:40

SPECIAL TEST 86-15

INTAKE COOLING WATER SYSTEM FLOW TEST

Background Information:

The purpose of this test is to obtain the flow rates and motor load of each ICW pump. The system configuration simulates the post LOCA alignment with the exception of no isolation of the Turbine Plant Cooling Water (TPCW) Heat Exchangers.

Test Results:

This Special Test was run on April 29, 1986. The results were evaluated by Engineering and the data from the test was used in the Emergency Diesel Generator load studies which were vital to the restart of Unit 4.

Safety Evaluation:

This test does not involve an unreviewed safety question because:

1. a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR: The test will be performed with Unit 4 in cold shutdown (Mode 5). Flow to the CCW Heat Exchangers will be maintained throughout the test duration. No ICW components required to maintain or provide additional flow to the CCW Heat Exchangers will be taken out of service. Therefore, if additional flow is required it can be provided as required and the probability of an accident previously evaluated in the FSAR will not be increased.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: Unit 4 will be in Mode 5 for the duration of the test. The ICW System will be aligned, with the exception of no isolation to the TPCW Heat Exchangers, in the post LOCA configuration. All essential ICW components will be operable, therefore, the severity of an accident previously evaluated in the FSAR would not be increased.

   c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: Throughout the duration of the test there will be monitoring of the motor amperage and the flows through each of the CCW and TPCW Heat Exchangers to ensure the limitations of each component will not be exceeded. High motor amperage would indicate a pump is approaching runout condition with an increasing NPSH requirement. The discharge valve from both TPCW Heat Exchangers will be throttled close to reduce the motor load. In the event of an ICW pump tripping, the same discharge valve will be throttled close prior to starting another ICW pump. During operation of two ICW pumps with the same discharge valve completely open, the discharge valve from each individual Heat Exchanger will be throttled close if the flow limitation of the applicable Heat Exchanger is approached.

The monitoring of the component parameters, limitations of each component, and the method of reducing the parameter is incorporated into the test procedure. Therefore, probability of malfunction of equipment important to safety will not be increased.

d. With respect to the consequneces of malfunction of equipment important to safety previously evaluated in the FSAR: Unit 4 will be in Mode 5, all ICW components important to safety will be operable and able to replace an active ICW component in the event of a single failure. The test does not involve manipulating equipment important to safety from other systems of the plant. The test procedure incorporates the limiting parameters, precautions and the method for reducing the parameters. Therefore, the consequences of malfunction of equipment important to safety will not be more severe.

2. a. With respect to the possibility of an accident of a different type than any analyzed in the FSAR: Unit 4 will be in Mode 5 throughout the duration of the test. The parameters associated with the unit in Modes above 5 are significantly reduced. The test does not involve manipulating components of systems important to safety not discussed in this text. The Precautions and Limitations for testing the ICW system are incorporated in the test procedure. Therefore, the possibility of an accident of a different type than any analyzed in the FSAR will not be increased.

   b. With respect to the possibility of malfunction of a different type than any analyzed in the FSAR: With the unit in Mode 5, no equipment from other plant systems not discussed in this text is manipulated, and the ICW system is operable. The possibility of malfunction of equipment of a different type than any analyzed in the FSAR will not be increased.

3. With respect to the margin of safety as defined in the Bases for a Technical Specification (T.S.): The ICW system will be maintained operable in accordance with the T.S. 3.4-5 and the margin of safety as defined in the Bases for a T.S. will not be reduced.

SPECIAL TEST 86-16

ADDITIONAL EMERGENCY DIESEL GENERATOR LOADING DATA COLLECTION

Background Information:

The purpose of this Special Test is to simulate CCW systems flow and CCW pump motor loading which would occur in the injection phase of a LOCA and to obtain kw loading data for the diesel JCO evaluation.

Test Results:

The test was run on May 3, 1986. The results were evaluated by Engineering and the data from the test was used in the Emergency Diesel Generator load studies which were vital to the restart of Unit 4.

Safety Evaluation:

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

1. a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR: No changes are being made that would increase the likelihood of an accident. This test will be performed while Unit 4 is in a cold shutdown condition (Mode 5). This test simulates the CCW system configuration during the Injection phase of a LOCA and, therefore, will provide confirmation that the system meets the required EDG loading requirements set forth in the FSAR and in JPE-L-86-59, Rev. 1.

   b. With respect to the consequences of an accident previously evaluated in the FSAR: The safety-related loads which require flow from the CCW system while in Mode 5 are the Spent Fuel Pit (SFP) heat exchangers, RHR heat exchangers and RHR pump seal water heat exchangers.

   c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR: The probability of malfunction of equipment will not be increased since no equipment will be modified. During the test, the CCW system will be aligned and operated within its design basis configuration. If the CCW pump being tested should fail, a second CCW pump could be started, if not already running, from the Control Room to maintain flow. As described in 1.b above, the safety-related loads which require cooling during the test are the Spent Fuel Pit heat exchanger, RHR heat exchanger, and RHR pump seal water heat exchangers. Flow will not be present in the RHR heat exchangers so long as the RCP and Steam Generator coolant loop remains operable, however, the RHR heat exchangers will be available if required. If the CCW pump being tested should fail, a second CCW pump could be started, if not already running, from the Control Room to maintain flow, and there exists adequate time to re-establish CCW flow to maintain the normal operating temperature of the Spent Fuel Pit and the RHR pump seal water within its temperature limit and the RCS temperature within the Mode 5 Technical Specifications limit.

d. With respect to the consequences of malfunction of the equipment important to safety previously evaluated in the FSAR: During the test, the CCW system will be aligned and operated within its design basis limitations. Therefore, the consequences of malfunction of the equipment important to safety will be no different from that previously evaluated in the FSAR. As described in 1.c above, in the unlikely event of a CCW pump failure, at least one (1) of the two 100% capacity standby CCW pumps will be available to maintain system operation, if the second pump is not already running.

2. a. With respect to the possibility of an accident of a different type other than any analyzed in the FSAR: This test simulates the configuration and operation of the CCW system which would exist during the injection phase following a LOCA so that additional data can be obtained for the emergency diesel generator loading.

The test will be performed while Unit 4 is in a cold shutdown condition (Mode 5). In the unlikely event of a CCW pump failure, there is sufficient time to place another CCW pump in operation, if an additional pump is not already running, to provide flow to the SFP heat exchanger, RHR heat exchanger (if required), and the RHR pump seal water heat exchangers.

Based upon the preceeding, the possibility of an accident of a different type other than any analyzed in the FSAR will not be created.

b. With respect to the possibility of a malfunction of a different type other than any analyzed in the FSAR: The proposed test simulates CCW system alignment and performance which would exist during the injection phase after a LOCA, but does so in a controlled manner under the direction of a dedicated Test Coordinator with support from Plant Operations staff including direct communication with the Control Room Operators. Therefore, the possibility of a malfunction of a type different than any previously evaluated in the FSAR will not be created.

3. With respect to the margin of safety as defined in the basis for any Technical Specifications: The margin of safety will not be decreased because the test will be performed during a cold shutdown condition. In addition, adequate precautions are noted in the test procedure which will assure that all applicable Technical Specifications limits will be maintained during the performance of this test.

SPECIAL TEST 86-17

INTAKE COOLING WATER SYSTEM FLOW TEST - UNIT 3

Background Information:

This test was performed to measure total Intake Cooling Water flow and motor loads from one or two Intake Cooling Water Pumps with combinations of two or three Component Cooling Water Heat Exchangers and both Turbine Plant Cooling Water Heat Exchangers with the Isolation Valve to the Turbine Plant Cooling Water Heat Exchangers fully open.

Test Results:

This test was performed on May 9, 1986, and the test results were forwarded to Engineering for evaluation and review.

Safety Evaluation:

The performance of this test does not involve an unreviewed safety question nor does it increase the probability of an accident because:

The performance of this test does not require any system changes which would increase the probability of occurrence of an accident. This test will be performed while Unit 3 is in Power Operation. The first portion of this test simulates the Intake Cooling Water System configuration following a LOCA with a postulated failure to isolate the Turbine Plant Cooling Water (TPCW) Heat Exchangers. The purpose of this test is to verify that any one (1) Intake Cooling Water (ICW) pump will provide the post-LOCA required flow through the Component Cooling Water (CCW) Heat Exchangers, assuming flow through the Turbine Plant Cooling Water Heat Exchangers is not isolated.

The second portion of this test simulates the Intake Cooling Water system configuration when Unit 3 is in power operation. The purpose of this test is to measure the flow from two ICW pumps with the Isolation Valve to the TPCW Heat Exchangers fully open and combinations of two and three CCW Heat Exchangers in service.

The consequences of any accident previously evaluated in the FSAR will not be increased, since all three ICW and all three CCW pumps will be operable for the duration of the test. Both portions of the test will monitor temperature of the TPCW and CCW Heat Exchangers shell side effluents, so that the design limits are not exceeded. In addition, the ICW system will be maintained operable in accordance with Technical Specification 3.4-5 for power operation, throughout the course of the test. Finally, a caution is included in the procedure which requires an operator to open the ICW Isolatio Valve in the event of a SI during the test.

The probability of a malfunction of equipment important to safety will not be increased based on incorporation of the following precautions into the test procedure:

1. Minimum required pump submergence will be ensured by verification of Intake Canal level before any ICW pump is started.

2. Lube water flow to each ICW will be maintained within specified limits.

3. ICW pump motor current will be monitored, if normal ICW pump motor current is exceeded, which could indicate pump runout, flow from the TPCW Heat Exchangers will be reduced by throttling the discharge valve.

4. In the event of excessive vibration, unusual noise, overheating or other abnormal symptoms, the test will be discontinued and corrective action taken.

The precautions listed above are intended to prevent pump damage due to pump runout, loss of lube water or operation with NPSH available less than NPSH required.

Flow rates through the CCW and TPCW heat exchangers will be maintained below the maximum specified limits to prevent vibration induced tube damage.

If required, the ICW flow in the CCW heat exchangers may be increased to the units described in letter JPE-PTPO-86-522. Increased flow rates through these heat exchangers will be limited to minimum periods of time.

The first portion of the test aligns a single ICW pump with the TPCW and CCW Heat Exchangers, this is not the normal power operation configuration However, as stated above, adequate precautions have been incorporated into the procedure to preclude pump damage as well as other abnormalities such as heat exchangers outlet temperatures. Throughout the test, with non-operating pumps will be secured. In the unlikely event of an ICW pump failure, the test would be terminated and the standby pumps started to maintain system operability.

The first poriton of this test simulates the Intake Cooling Water System configuration following a LOCA with a postulated failure to isolate the Turbine Plant Cooling Water heat exchangers. Data obtained from this test will be used to verify that any one (1) ICW pump will provide the post-LOCA required flow through the CCW heat exchangers assuming flow through the TPCW heat exchangers is <u>not</u> isolated.

The test will be accomplished in a controlled manner in accordance with an approved procedure under the direct supervision of a dedicated Test Coordinator who will be in communication with the Control Room Operators. As such, system and equipment parameters will be monitored closely, with appropriate actions taken to prevent component damage and return the system to normal operating limits.

Based on the preceeding, the possibility of an accident of a different type than any analyzed in the FSAR will not be created.

The test will be run with combinations of one and two ICW pumps together with two and three CCW heat exchangers. The TPCW heat exchangers will remain in service throughout the test. This test is performed in a controlled manner under the direction of a dedicated Test Coordinator with support from plant operations staff including direct communication witht the Control Room Operators. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the FSAR will not be created.

The margin of safety will not be decreased because the ICW system will be maintained operable in accordance with Technical Specification 3.4-5 throughout the course of the test.