ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9009100024 DOC.DATE: ~~90/08/30~~ 8/30 NOTARIZED: NO DOCKET #
FACIL:50-250 Turkey Point Plant, Unit 3, Florida Power and Light C 05000250
50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
AUTH.NAME AUTHOR AFFILIATION
HARRIS,K.N. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

See Report

SUBJECT: "Changes,Tests & Experiments Made as Allowed by 10CFR50.59 for Period of Jul 1989 - June 1990."

DISTRIBUTION CODE: IE47D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 144
TITLE: 50.59 Annual Report of Changes,Tests or Experiments Made W/out Approv

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 LA	1	0	PD2-2 PD	5	5
	EDISON,G	1	0			
INTERNAL:	AEOD/DOA	1	1	AEOD/DSP/TPAB	1	1
	~~NRR/DLPQ/LHFB11~~	1	1	NRR/DOEA/OEAB11	1	1
	REG FILE 02	1	1	RGN2 FILE 01	1	1
EXTERNAL:	NRC PDR	1	1	NSIC	1	1

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 15 ENCL 13

MA 14 uw

P.O. Box 14000, Juno Beach, FL 33408-0420

FPL

AUG 30 1990

L-90-306
10 CFR 50.59

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

Gentlemen:

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

Florida Power & Light Company's Report on "Changes, Tests and Experiments Made Without Prior Commission Approval" for the period July 1, 1989 through June 30, 1990 is attached.

Very truly yours,

KN Harris

K. N. Harris
Vice President
Turkey Point Plant Nuclear

KNH/DRP/MKA/mka

Attachment

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

IE47
1/1

9009100024

ANNUAL
10 CFR 50.59
REPORT

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 & 4

TURKEY POINT PLANT UNITS 3 AND 4

DOCKET NUMBERS 50-250 AND 50-251

CHANGES, TESTS, AND EXPERIMENTS

MADE AS ALLOWED BY 10CFR50.59

FOR THE PERIOD OF

JULY 1, 1989 THROUGH JUNE 30, 1990

INTRODUCTION

This report is submitted in accordance with 10CFR50.59(b), which requires that:

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

which are conducted without prior Commission approval be reported to the Commission at least annually. This report is intended to meet this requirement for the period of July 1, 1989 through June 30, 1990.

This report is divided into five (5) sections; the first, changes to the facility as described in the SAR performed by a Plant Change/Modification (PC/M); the second, changes to the facility or procedures as described in the SAR not performed by a PC/M and tests and experiments not described in the SAR; the third, a summary of the Unit 3 cycle 12 reload evaluation; the fourth, a list of Power Operated Relief Valve (PORV) actuations, which is submitted in accordance with FPL's commitment to comply with the requirements of Item IIK.3.3 of NUREG 0737; the fifth, a summary of the findings of the Unit 3 Steam Generator tube inspection.

Table of Contents

SECTION 1

Changes to the facility as described in the SAR performed by a PC/M.

PLANT CHANGE/MODIFICATION 83-217

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	03/12/90

BORIC ACID EVAPORATOR PUMP UPGRADE

Summary:

The function of this modification was to replace the Unit #4 Boric Acid Evaporator feed tank pumps previously installed under PC/M 77-11. The current pumps require approximately 9 feet Net Positive Suction Head (NPSH) while only 1 to 2 feet NPSH is currently available. The replacement pumps require only 2 feet NPSH and were installed at an elevation to maximize NPSH available.

Safety Evaluation:

This modification does not involve an unreviewed safety question because of the following:

1. The boric acid evaporators performed no safety function and are classified non-nuclear safety related.

2. This modification does not interact with any safety related systems or components.

3. No safety related equipment or components are compromised by any assumed failure of any existing or new equipment or components.

4. The pressure integrity of the new pumps and piping is equal to those components that are part of the original installation.

5. The foundation for the boric acid concentration pumps does not perform a nuclear safety related function and will not adversely affect any safety related systems or equipment. No new drilling into the existing Class 1 floor slab is required and the additional weight of the new pump is within the design live loads for the floor slab.

6. The pipe supports for the boric acid system do not perform a nuclear safety related function and are located

away from safety related equipment and components.

7. The Boric Acid Evaporator System is non-safety related. All cables are run in non-safety related raceways. No cables are routed in cabletrays, therefore the current capacity of any cable in the raceway is not affected. The motor starters and panel breaker are non-safety related and are fed from a non-vital MCC.

8. The heat tracing system only provides heat to non-nuclear safety related pumps, therefore it does not affect any safety systems.

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, previously evaluated in the FSAR, has not been increased. There is no possibility of accident or malfunction different than those previously evaluated. Also, there are no changes required to the Technical Specifications of the plant. Therefore, it can be concluded that the boric acid concentration pump foundation does not pose an unreviewed safety question pursuant to 10CFR 50.59.

PLANT CHANGE/MODIFICATION 84-109

PC/M CLASSIFICATION	:	NSR
UNIT	:	3 & 4
TURN OVER DATE	:	08/15/89

SITE PREPARATION FOR PLANT EXPANSION

Summary:

This PCM was issued for the placement of backfill south of Turkey Point Units 3 and 4. The placement of this fill is required for the planned expansion of the Turkey Point Plant site and the addition of new buildings and other facilities south of Unit 4 in support of the Turkey Point Site Facilities Upgrade Program. All work associated with this PCM is not safety related. The fill will be obtained from existing stockpiles of limerock fill located along the site discharge canal system.

Safety Evaluation:

No nuclear safety related systems are affected or associated with these modifications. No safety related facilities or structures are planned to be installed in the new fill area under this PCM. The modifications covered by this PCM are not inside containment, do not involve safety related snubbers, do not affect the spent fuel cooling operation of the plant, and do not affect radioactive waste treatment systems or plant effluent. Also, any new cable will not be routed inside original plant raceways. No new equipment will be attached to or in proximity to block walls classified as safety related under I.E. Bulletin 80-11.

PLANT CHANGE/MODIFICATION 85-030

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	01/31/90

4160 SWITCHGEAR SYNCHRONIZING CIRCUIT MODIFICATIONS

Summary:

As presently connected the contact circuit of synchronizing check relays 125/4A, 125/4B and 125/4CBT (G.E. type IJS) for 4160 V busses 4A, 4B and 4C respectively is continuously loaded which can result in contact welding. This contact is used to energize an auxiliary relay 125X/4A (B) or (C) in a DC circuit which in turn permits the closing of the respective 4160 Volt incoming circuit breaker. This modification utilized a 'close' contact of the circuit breaker control switch in series with the synchronizing check relay contact so that this contact carries the load current only during the manual synchronizing sequence. This change added only one half cycle to total breaker close time.

Safety Evaluation

The modification does not change the basic operation of the synchronizing sequence and does not add new operating modes to the transformer breakers. Therefore the possibility of malfunction of a different type than analyzed in the FSAR is not created.

The modification does not alter the function of the synchronizing check relay or affect the transformer breaker's operation and does not increase the probability of occurrence or the consequences of an accident or equipment malfunction. On the contrary, it prevents the synchronizing check relay contact from possibly being welded. The margin of safety defined in the bases for Technical Specification 3-7 will not be affected. For these reasons, no reduction in any Technical Specification margin of safety will result.

PLANT CHANGE/MODIFICATION 85-066

PC/M CLASSIFICATION	:	SR
UNIT	:	3 & 4
TURN OVER DATE	:	06/15/90

REMOVAL, REPLACEMENT & MODIFICATION OF CONTROL ROOM DOOR

Summary:

PC/M 85-066 provided for the purchase and installation of a new entry door fitted with a new cast steel lockset. The existing door hinges were also replaced with new hinges. After removal, the existing door was modified by replacing the existing lockset with a new reinforced lockset, same as for the new door, and will be used as a spare. The alternate existing door is to be used as a standby replacement door.

Safety Evaluation

Modifications under this PCM are not attached to or in the vicinity of safety related block walls, do not affect radioactive waste treatment systems, or plant effluent, and do not involve any safety related pipe snubbers or additional cables.

This PC/M does not involve modifications in the containment, therefore, the ECCS Heat Sink Analysis is not affected.

PLANT CHANGE/MODIFICATION 85-116

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	10/16/89

ADDITIONAL CABLE REROUTING FOR APPENDIX R MODIFICATIONS

Summary:

This PCM Design Package provided design changes to meet 10CFR50 Appendix R, Section III.G, "Fire Protection of Safe Shutdown Capability Requirements," as interpreted by FPL in letter JPE-PTPM-85-2038, dated November 29,1985. Cables covered by this PCM are associated with Unit 3 equipment and are in addition to those covered by PCM 83-151.

To arrive at the circuits requiring reroute, a C-Bus and Standby Steam Generator Feedwater System analysis was completed. Additionally, it was determined that a number of cables previously identified as requiring manual actions during a fire, are now being rerouted to enhance the plants ability to safely shutdown during the postulated fire.

Safety Evaluation

Since these changes consist only of the physical relocation of items and the pulling and termination of new qualified cable, they do not change the control of any electrical equipment and do not change the limiting conditions for operation of the plant Electrical Systems as described in Technical Specification 3.7 or the principle design criteria defined in Section 8.1 of the Turkey Point FSAR.

PLANT CHANGE/MODIFICATION 85-142

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	08/03/89

FUEL TRANSFER SYSTEM MANIPULATION CRANE DUAL CABLE MODIFICATIONS

Summary:

This Engineering Package covered modifications to the Unit 4 manipulation crane. These modifications consist of upgrading the manipulator crane from a single hoist to a dual cable hoist along with the installation of a new hoist load indicator system and the installation of a load test fixture which will be used to load test the main hoist on the manipulator cranes.

Implementation of this package adds redundancy to the cable hoist portion of the refueling machine. This enhancement increased reliability, reduces maintenance time, and incorporates the latest state of the art capabilities into the refueling system.

Safety Evaluation

The electrical modifications consist of the replacement of a single cable hoist with a dual cable hoist motor, the replacement of mechanical geared limit switches with programmable limit switches, and replacement of the Dillon load sensing device with a Sensotec device. The addition of the above equipment does not adversely affect any safety related equipment as the equipment is located on the manipulator crane and is fed from a non-safety related power supply. All electrical equipment installed by this PCM has been installed seismically to prevent interaction with any safety related equipment during a seismic event.

Based on the above, it can be demonstrated that an unreviewed safety question, as defined by 10CFR 50.59 does not exist and that no Technical Specifications are affected.

PLANT CHANGE/MODIFICATION 85-148

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	11/30/89

SPENT FUEL POOL COOLING SYSTEM - SEISMIC UPGRADE

Summary:

This Engineering Package provided the design to upgrade the Spent Fuel Pool (SFP) Cooling System to ensure that the cooling function of the system is not lost as a result of a seismic event. Also included in this package was the upgrading of the system piping to include pool boiling (212 degrees F) as the operating temperature for piping stress analysis purposes only.

Several modifications are required in order to accomplish the system upgrade. First, a thermal expansion loop is added to the existing piping in order to reduce the thermal loads on the pump nozzles. Secondly, several existing pipe supports are added in order to accommodate seismic loads. Finally, the manual transfer switch for the Spent Fuel Pool pumps was replaced with a seismically qualified switch.

Safety Evaluation

There are no changes to the Turkey Point Technical Specifications due to this modification. The SFP cooling System is not include in any bases for Technical Specifications and does not adversely impact any Safety Related system. Therefore, there is no effect to any Technical Specification due to this modification.

PLANT CHANGE/MODIFICATION 85-150

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	07/17/89

SPENT FUEL POOL (SFP) - AIR INLET DAMPER REPLACEMENT

Summary:

This Engineering Package provided for the replacement of the two existing air inlet dampers, integral roughing filters, and damper actuators for the Spent Fuel Pool Ventilation System. The replacement dampers and actuators purchased by FPL are pathway Parallel Multiblade Dampers provided with integral filters.

The replacement dampers, integral filters, and damper actuators restored the Spent Fuel Pool Ventilation System to its originally intended design configuration. Modification to the power supply, control circuits, and electrical raceways associated with the new damper configuration and increased power requirements.

Safety Evaluation

This installation does not adversely affect the integrity, operation, or function of any safety related system addressed in the existing or proposed Technical Specifications. Therefore, the margin of safety, as defined in the bases for the Plant Technical Specifications, is not decreased.

This EP does not constitute an unreviewed safety question. Thus, prior NRC approval was not required.

PLANT CHANGE/MODIFICATION 86-83

PC/M CLASSIFICATION	:	NNS
UNIT	:	3 & 4
TURN OVER DATE	:	10/18/89

NUCLEAR MAINTENANCE BUILDING

Summary:

This engineering package addressed the renovation of the Turkey Point Nuclear Maintenance Facility. In order to upgrade and consolidate maintenance facilities and provide for the effective utilization of maintenance manpower, it was decided to remove the existing Nuclear Maintenance Building and construct a new, larger facility on the existing building foundation.

The original Nuclear Maintenance Building (NMB) was a pre-fabricated metal structure on a reinforced concrete slab and grade beam foundation. The structure measured approximately 60 feet wide by 215 feet long. The building was used for personnel access control to the Radiation Control Area, construction offices and light storage.

The new structure is a reinforced concrete and masonry structure. The exterior architectural treatments match those of the adjacent Nuclear Administration Building (NAB). The first floor finish floor level was raised approximately 18 inches to attain compatibility with the adjacent Nuclear Administration Building.

Safety Evaluation:

The Nuclear Maintenance Building and the crossover bridge will not add to, modify or affect any plant nuclear safety related systems nor will it perform a plant nuclear safety related function. None of these systems has a leakage to plant nuclear safety related systems.

The addition of the Nuclear Maintenance Building and the addition of the crossover bridge does not pose an unreviewed safety question.

PLANT CHANGE/MODIFICATION 86-127

PC/M CLASSIFICATION : NNS
UNIT : 3
TURN OVER DATE : 06/20/90

REACTOR COOLANT PUMP OIL COLLECTION SYSTEM MODIFICATION/ANALYSIS

Summary:

This engineering package covered the modification to the Reactor Coolant Pump (RCP) Oil Collection System piping. This modification installed flexible hoses between the pumps and the hard piping which drains the oil to the oil collection tank. Installation of the flexible hoses provides free thermal movement of the RCP and aids in the removal of the RCP motors for repair or replacement. This package also provides for the seismic analysis and includes upgrade of the RCP Oil Collection System to meet the requirements of 10CFR50 Appendix R Section III.O.

Safety Evaluation:

This modification does not affect the FSAR or require a change to the plant Technical Specifications. There are no unreviewed safety questions.

PLANT CHANGE/MODIFICATION 86-137

PC/M CLASSIFICATION	:	NSR
UNIT	:	4
TURN OVER DATE	:	08/19/89

MAIN STEAM SAFETY VALVES - LIFTING DEVICES

Summary:

This Engineering Package (EP) removed lifting devices from the Main Steam Safety Valves (MSSV'S) RV1400, 1401, 1402, 1403, 1405, 1406, 1407, 1408, 1410, 1411, 1412 AND 1413. Removal of these devices will preclude failure of the MSSV's to reseat due to malfunction or loss of any component of the lifting devices.

Safety Evaluation:

The modification included in this Engineering Package is considered to be safety related and does not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated is not increased since the removal of lifting devices from the MSSV's will not impact the intended operation of these valves. Rather, the modification will minimize malfunction of the MSSV's which could happen due to failure of the lifting devices resulting in excessive heat removal from the steam generator caused by struck open MSSV.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no changes have been made to the operational design of the MSSV's.

(iii) This modification does not change the margin of safety as defined in the bases for any technical specification.

Implementation of this PCM does not require a change to the plant Technical Specifications. This change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

PLANT CHANGE/MODIFICATION 86-154

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 03/14/90

ISOLATION FOR SV-20-OPC: TURBINE OVERSPEED PROTECTION SYS CTRL OIL DUMP VLVS

Summary:

This Engineering Package (EP) is to provide isolation valves on either side of two (2) solenoid valves to be tagged SV-20-0PC-1 and 2 to allow for maintenance during plant operation. The valves I SV-20-0PC-1 and 2 are part of the Turbine Overspeed Protection system. On a turbine overspeed signal, these valves open and dump the system control oil, thereby closing the Main Steam control and intercept valves. This action in anticipation of a speed change prevents the turbine from overspeeding. Additionally, the solenoid valves were replaced by new identical model solenoid valves, due to wear and failures caused by aging.

Safety Evaluation:

The modification included in this Engineering Package is considered to be Quality Related and does not involve an unreviewed safety question. The following are the bases for this justification:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. The only equipment associated with this modification that was considered in a safety analysis was the turbine generator set itself as a potential source of a missile in an overspeed event. This analysis is incorporated in the FSAR, Appendix SE. However, Section 14.1.13 of the FSAR states that due to the level and frequency of inspection and vendor analysis, turbine generator missiles are no longer considered a credible postulated event. Additionally, Appendix 5E has already assumed turbine missile generation due to overspeed and as a result any Safety Related features affected have been provided protection from this unlikely event. Additionally, as discussed in 2.1.1, numerous backup overspeed trip devices are available to provide a turbine trip which precludes the potential for turbine missiles and results in a reactor trip. As such, this equipment is quality related and will have no effect on equipment

vital to plant safety.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PC/M, since the components involved in this modification are not included in the bases of any Technical Specification.

PLANT CHANGE/MODIFICATION 86-181

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	04/23/90

SI PUMP MINI-RECIRCULATION VALVE ACTUATOR REPLACEMENT

Summary:

This engineering package details the replacement of the actuators on Isolation Valves 3-856A and 3-856B for the Safety Injection (SI) pump minimum flow recirculation lines.

Valves 3-856A and 3-856B are pneumatically operated and in series. If the valves lose power or instrument air, they fail closed. During a small break loss of coolant accident, the RCS pressure may be greater than the SI pump discharge pressure for a short length of time. If either Valve 3-856A or Valve 3-856B fail closed, then the SI pumps do not have a minimum flow recirculation path which could result in pump damage.

This modification replaced the existing air operators with motor operators to eliminate potential single failure concern. The MOV control circuit will incorporate the use of a separate control power "on-off" switch in addition to the normal valve control switch. The control power switch will be used to "isolate" the MOV control circuit from control power whenever the valve is not being operated. This method of MOV control circuit "isolation" will enable the MOV to accommodate an electrical short single failure which could create a potential for undesired valve closure. Since these valves will be open during normal operation and the SI injection phase, minimum flow recirculation will be available in the event of a single failure. Isolation of the RWST will also be available in the event of a single failure since the valves are in series and only one valve is required for isolation. The valves will also be equipped with a handwheel for backup operation.

Safety Evaluation:

This modification involves replacing the actuators on Valves 3-856A and 3-856B which are located in the CCW pump and HX area. This modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. This modification does not constitute an unreviewed safety question. Therefore, prior NRC approval of this EP was not required.

This page was intentionally left blank.

PLANT CHANGE/MODIFICATION 86-190

PC/M CLASSIFICATION	:	QR
UNIT	:	3 & 4
TURN OVER DATE	:	02/20/90

EDG STARTING AIR SYSTEM AIR QUALITY IMPROVEMENT

Summary:

This Engineering Package provided air quality improvements to the Emergency Diesel Generator (EDG) starting air system.

NUREG/CR-0660 states that water in the starting air system either directly or indirectly is the "root cause" of many emergency diesel generator engine failures. NUREG/CR-0660 further states that draining the air reservoirs of condensate and using air strainers and filters is only partially effective. Dust and dirt have also proven to be a problem with the EDG starting air system.

This modification included the installation of air dryers, aftercoolers, and prefilters to improve the quality of the EDG starting air system.

Safety Evaluation:

The EDG Starting Air System air quality improvement equipment does not perform a Safety Related function, is not required to maintain the reactor coolant system pressure boundary, and does not adversely impact plant safety or operations. However, the equipment and associated piping are seismically supported to preclude potential interactions with Safety Related equipment. Therefore, the EDG Starting Air System is classified as Quality Related. The modification associated with this EP has no adverse effect on safety related components or systems, and does not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval to implement this change was not required.

PLANT CHANGE/MODIFICATION 86-194

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	10/03/90

ADDITION OF CONTINUOUS TUBE CLEANING CAPABILITY TO THE CCW HEAT EXCHANGERS

Summary:

This Engineering Package covered modifications for testing and evaluation to the Intake Cooling Water System to provide on-line mechanical tube cleaning for each CCW heat exchanger. The new cleaning systems will operate by introducing sponge rubber balls into the cooling water supply line of each heat exchanger. The normal process flow will then force the balls through the heat exchanger tubes, wiping them clean. Screens in the discharge lines will collect the balls and a centrifugal pump will recirculate the balls to the injection point. A ball collector is also included to allow addition or retrieval of the cleaning balls.

Safety Evaluation:

The modifications provided by this Engineering Package will have no impact on Plant safety and operation. The modifications for testing and evaluation do not constitute an unreviewed safety question, do not require changes to the Technical Specifications and, therefore, prior NRC approval was not required.

PLANT CHANGE/MODIFICATION 86-202

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	06/19/90

HYDROGEN DETECTION INSIDE EXCITER HOUSING

Summary:

This engineering package covered the addition of a hydrogen detection system mounted on the turbine-generator exciter and coupling housings. The purpose of the hydrogen detection system is to detect a hydrogen gas buildup in the housings before the gas concentration reaches the Lower Explosive Limit (LEL) and to alert the operators of this situation.

To perform this function, two detectors were installed on the turbine-generator assembly. One was placed directly on the exciter housing at the generator end and the other was located on the generator end removable cover above the exciter coupling. The two control units were mounted together in a NEMA 4 box at elevation +21' in the turbine building, out of the weather. The box includes a transparent front window which will enable operators to observe the meter and light indications on the control units without opening the box.

Safety Evaluation:

The modifications provided by this Engineering Package will have no impact on Plant safety and operation. The modifications for testing and evaluation do not constitute an unreviewed safety question, do not require changes to the Technical Specifications and, therefore, prior NRC approval was not required.

PLANT CHANGE/MODIFICATION 86-222

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	08/25/90

INSTALLATION OF FLANGES ON RCP SEAL LINES

Summary:

This Engineering Package covered modifications to the RCP No. 1 and No. 2 Seal Leak-off Lines and the No. 1 Seal Bypass line. Additional flanges were installed on the RCP Seal Leak-off and Bypass Lines to permit removal of spool pieces during maintenance of the RCP motors.

Safety Evaluation:

Detailed piping and support analyses were performed to determine the required system modifications and demonstrate the acceptability of the new configuration to FSAR criteria. Since no new active piping components were introduced and the piping and support analysis were performed in accordance with existing Design Criteria, the Margin of Safety as established in the FSAR has not been reduced and the probability of occurrence and consequences of an accident has not been increased. Also, the possibility of an accident other than previously postulated has not been introduced. In conclusion, the changes proposed in this design package do not involve any unreviewed safety questions and will not require changes to any plant Technical Specifications. Consequently, prior NRC approval for the implementation of this modification was not required.

PLANT CHANGE/MODIFICATION 87-285

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	05/01/90

REPLACEMENT OF TURBINE RUNBACK SELECTOR SWITCH

Summary:

This Engineering Package (EP) provided the design necessary to replace the temporary Turbine Runback Selector Switch with a permanent keylocked selector switch. This modification was required to complete the design associated with the Turbine Runback Modification initiated by PCM 84-210.

Safety Evaluation:

This modification does not constitute an unreviewed safety question or require changes to Plant Technical Specification; therefore, NRC approval was not required prior to implementation.

PLANT CHANGE/MODIFICATION 87-310

PC/M CLASSIFICATION : QR
UNIT : 3 & 4
TURN OVER DATE : 10/31/89

UPGRADE OF THE EMERGENCY RESPONSE DATA ACQUISITION AND DISPLAY SYSTEM

Summary:

The enhancement of the Emergency Response Data Acquisition and Display System (ERDADS) is a result of startup testing and debugging of the computer software. This Engineering Package provided for the installation of the updated computer and peripheral equipment to support software revisions provided by Energy Incorporated (EI).

Safety Evaluation:

This modification does not constitute an unreviewed safety question or require changes to Plant Technical Specification; therefore, NRC approval was not required prior to implementation.

PLANT CHANGE/MODIFICATION 88-026

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	10/12/89

PERMANENT SFP VISUAL LEVEL MEASURING SCALE

Summary:

This Engineering Package provided for the addition of a permanent visual water level scale for the Spent Fuel Pool. This scale will improve personnel safety by providing an alarm and will also improve the accuracy of local pool level measurement, without affecting the accuracy of the existing pool level transmitter to which the new scale will be attached.

Safety Evaluation:

A safety evaluation has been performed in accordance with 10CFR 50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question. Furthermore, the implementation of this modification does not require a change to the plant technical specifications and has no detrimental effects on plant safety and operation. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-027

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 10/12/89

PERMANENT SFP VISUAL LEVEL MEASURING SCALE

Summary:

This Engineering Package provided for the addition of a permanent visual water level scale for the Spent Fuel Pool. This scale will improve personnel safety by providing an alarm and will also improve the accuracy of local pool level measurement, without affecting the accuracy of the existing pool level transmitter to which the new scale will be attached.

Safety Evaluation:

A safety evaluation has been performed in accordance with 10CFR 50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question. Furthermore, the implementation of this modification does not require a change to the plant technical specifications and has no detrimental effects on plant safety and operation. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-097

PC/M CLASSIFICATION : QR
UNIT : 3
TURN OVER DATE : 05/24/90

CV-2011 VALVE STROKE CLOSING TIME (LP HEATER BYPASS)

Summary:

The Feedwater Heater Bypass Valve CV-2011 opens to provide condensate flow to the Steam Generator Pump (SGFP) when the suction pressure at the SGFP drops below the pressure switch setpoint of 220 psig. This is to maintain sufficient Net Positive Suction Head (NPSH) on the SGFP. However, a rapid closure of this bypass valve can cause a decrease in the suction pressure due to slow acceleration of the stagnant condensate through the low pressure feedwater heaters. This could potentially trip the SGFPs which could result in tripping of the reactor.

To eliminate this problem, this Engineering Package (EP) provided modifications to the pneumatic controls of Valve CV-2011. The installation of an air-flow metering valve in the actuator pneumatic control line reduced the instrument air flow rate on valve closure, thereby lengthening the CV-2011 bypass valve closure time.

Safety Evaluation:

Plant operation, security and safety will not be affected during or by the installation of this modification. In accordance with 10CFR50.59, the modification was made without prior NRC approval since the proposed change does not involve an unreviewed safety question and does not require a change to Plant Technical Specifications.

PLANT CHANGE/MODIFICATION 88-163

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	06/13/90

AFW NITROGEN BACKUP SYSTEM RELIEF VALVE INSTALLATION

Summary:

This Engineering Package (EP) provided the design for the replacement of existing Rupture Disc Assemblies on the AFW Nitrogen Backup Systems with dual mounted relief valves. The replacement relief valves were procured QL-1, were seismically qualified and were seismically installed. The function of these valves is to protect the AFW nitrogen backup supply instrumentation from over-pressure should the manifold pressure regulator fail open. Each AFW nitrogen backup system manifold has two new relief valves, mounted in a parallel sub-assembly, and installed in the same location as the replaced rupture disc.

In addition, this EP provided for the installation of six test valves located on each instrument air header to the AFW Flow Control Valves, between the isolation valve and the check valve for Unit 3. The primary purpose of these test valves was for leakage testing of the instrument air check valves.

Safety Evaluation:

Plant operation, security and safety was not be affected during or by the installation of this modification. In accordance with 10CFR50.59, the modification proposed by this EP was made without prior NRC approval since the proposed change does not involve an unreviewed safety question and did not require a change to Plant Technical Specifications.

PLANT CHANGE/MODIFICATION 88-196

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	05/07/90

INADVERTENT ACTUATION OF PRMS RELAYS R-11 AND R-12

Summary:

This engineering package (EP) modified the Unit 3 PRMS cabinet (3QR66) by installing terminal blocks, fuse blocks, and fuses, allowing each cabinet drawer to have a separately fused power supply and a separate system ground connection. Metal-oxide varistors were placed across external relay coils in 3QR66 to provide noise protection from voltage transients. Bypass switches were installed in Main Control Board 3C06, providing isolation of the actuator signals from channels R-3-11 and R-3-12 as an aid for maintenance, testing, and surveillance activities.

Safety Evaluation:

This change did not affect the Technical Specification and did not involve an Unreviewed Safety Question pursuant to 10CFR 50.59.

PLANT CHANGE/MODIFICATION 88-238

PC/M CLASSIFICATION	:	NSR
UNIT	:	4
TURN OVER DATE	:	03/14/90

BOWSER TURBINE LUBE OIL CONDITIONER LEVEL CONTROL

Summary:

This Engineering Package provided for the design and installation of a pneumatic level control system for the Bowser Turbine Lube Oil Conditioner (4F4) on Unit 4. The modification utilized a pneumatic level transmitter and controller combination to maintain an intermediate level of lube oil in the conditioner per the manufactures's recommendation. The level controller has proportional and integral (reset) function capabilities which modulate control valve CV-4-2820 to maintain the desired operating level of lube oil.

SAFETY EVALUATION

This PCM did not require prior NRC approval for implementation since the modifications did not constitute an unreviewed safety question or require a change to Plant Technical Specifications. This modification does not adversely affect plant safety or operation.

PLANT CHANGE/MODIFICATION 88-243

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	02/01/90

INSTALLATION OF TERMINAL BOXES FOR CONNECTION OF PORTABLE CARD READERS AT CONTAINMENT ENTRANCES

Summary:

This Engineering Package provided the design and installation of terminal boxes at the containment personnel hatch and equipment hatch for connection of portable card readers. The use of card readers enables the members of the security force to verify access level for each individual prior to granting access into the containment building. The connectors in each terminal box were wired to the Plant Security System.

SAFETY EVALUATION

This modification does not constitute an unreviewed safety question nor does it require any changes to the Plant Technical Specification; therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 88-248

PC/M CLASSIFICATION : NSR
UNIT : 3 & 4
TURN OVER DATE : 08/18/89

MODIFICATIONS TO TELEPHONE SYSTEM MANHOLES

Summary:

Several communication manholes (installed under PC/M 85-121) required modification to accommodate vehicular traffic since these manholes were not flush with the existing grade. This Engineering Package provides the modifications required to allow passage through the parking areas and to facilitate vehicular access.

SAFETY EVALUATION

This modification does not involve an unreviewed safety question, and does not require any changes to the Turkey Point Technical Specifications; therefore, prior NRC approval for implementation of this modification was not required.

PLANT CHANGE/MODIFICATION 88-251

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	03/10/90

PROTECTIVE GUARDS FOR MG-6 SAFEGUARD RELAYS

Summary:

This Engineering Package provided the design for installation of protective guards for the MG-6 safeguard relays located in four safeguard racks. This protective guard installation was required to minimize inadvertent unit trips and Engineered Safety Feature challenges by shielding the relays from the cable bundles which connect interior-mounted components to door-mounted components. On at least two occasions, these cable bundles have caused unit trips by bumping the MG-6 relays during door movement. Therefore, any work to the interior of the safeguard rack has been inaccessible during normal plant operation. The new protective guards were seismically designed and installed, can be removed for maintenance and were designed to allow for manual reset of the relays by the operator without removal.

SAFETY EVALUATION

This modification does not constitute an unreviewed safety question or require changes to the Plant Technical Specifications; therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 88-265

PC/M CLASSIFICATION	:	QR
UNIT	:	3 & 4
TURN OVER DATE	:	02/12/90

OIL RETENTION BASIN FOR EMERGENCY FIRE PUMP DIESEL OIL STORAGE TANK

Summary:

This Engineering Package provided the design for installation of an oil retention basin for the Emergency Fire Pump Diesel Oil Storage Tank. The function of the oil retention basin is to retain any potential diesel oil spill from the existing 550 gallon storage tank to assure compliance with environmental requirements.

SAFETY EVALUATION

This modification does not constitute an unreviewed safety question or require changes to the Plant Technical specifications nor adversely affect plant operation and safety; therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 88-345

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	05/11/90

TURBINE PLANT COOLING WATER ISOLATION VALVE MODIFICATION

Summary:

This modification provided for the replacement of existing Turbine Plant Cooling Water (TPCW) manually operated isolation valves 3-50-314 and 334 and installation of Air Operated Valves POV-3-4882 and 4883, such that these valves will close on a Safety Injection Actuation Signal. Automatic isolation of TPCW heat exchangers during accident conditions ensures required Intake Cooling Water flow to the Component Cooling Water heat exchangers. Consequently, the modification resolves the single failure concern associated with Valve CV-3-2201.

SAFETY EVALUATION

This modification does not constitute an unreviewed safety question and does not require a change to the Technical Specifications. Also, this modification does not adversely affect Plant safety, security or operation. Therefore, implementation of this modification did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-514

PC/M CLASSIFICATION : QR
UNIT : 3
TURN OVER DATE : 02/10/90

AREA RADIATION MONITORING SYSTEM; CHANNEL 19 CABLE REPLACEMENT

Summary:

This Engineering Package (EP) provided the design documents, references and instructions for replacing a faulty Area Radiation Monitoring System (ARMS) cable between Control Room Rack 3QR30 and local indicator RI-1437 because of low insulation resistance in three of the eight conductors in the existing cable. The replacement cable along with three spare cables were coiled in 3QR30 and in a Pull Box located near RI-1437, and will be activated in the event of further ARMS cable failures.

The added raceways, replacement cable and three spare ARMS cables will not modify or affect the integrity, operation or function of any safety related structures, systems or components. The modification associated with the EP was designed in accordance with requirements of the UFSAR, Appendix 5A in order to preclude any interaction with safety related structures, systems, or components.

SAFETY EVALUATION

The modification provide by this Engineering Package has been reviewed and determined to have no adverse effect on plant safety or operability. This Engineering package does not involve an unreviewed safety question or require any change to the Plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 88-587

PC/M CLASSIFICATION	:	SR
UNIT	:	3 & 4
TURN OVER DATE	:	12/19/89

CONTROL ROOM VENTILATION SYSTEM TEMPERATURE CONTROLLER MANUAL OVERRIDE

Summary:

The Control Room Ventilation System was identified as being susceptible to potential single failures. The potential single failure of the temperature controller (TC-6548) was identified as requiring the installation of an alternate means to ensure continued compressor operation. A Temporary System Alteration was utilized to provide manual override of the temporary controller.

The modification provided by this Engineering Package is an interim installation until permanent modifications resolving single failure concerns are issued and implemented. This modification was provided to replace a currently installed Temporary System Alteration. In addition, this Engineering Package performed a drawing update to reflect the as-installed configuration.

SAFETY EVALUATION

This Engineering Package provided an interim installation which does not constitute an Unreviewed Safety Question nor require a change in the Plant Technical Specifications and is, therefore, acceptable in accordance with 10CFR50.59. The as-installed configuration is bounded by the original design and criteria for the Plant. Therefore, the modification performed by this Engineering Package was implemented without prior NRC approval and does not adversely affect plant safety or operation.

PLANT CHANGE/MODIFICATION 89-112

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	05/08/90

BORIC ACID BLEND FLOW CONVERTER RELOCATION

Summary:

This Engineered Package provided the design and instructions for the replacement and relocation of Boric Acid Blender Flow Converter FM-3-113. Originally, the flow converter was located approximately 494 feet away from flow transmitter FT-3-113 and was not powered from the same circuit as the transmitter. Signal interference due to long transmission distance, and powering of the transmitter and the converter from two different power circuits prevent the proper performance of the converter. Furthermore, the existing flow converter was obsolete and needed to be replaced by a current version.

The modification provided by this Engineering Package is an interim installation until permanent modifications resolving single failure concerns are issued and implemented. This modification is provided to replace a currently installed Temporary System Alteration. In addition, this Engineering Package performed drawing update to reflect the as-installed configuration.

SAFETY EVALUATION

The implementation of this modification does not change the functional or operational requirements of the existing system. There are no adverse effects on plant operation or plant safety, and this modification does not constitute an unreviewed safety question nor does require changes to Plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-128

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	03/14/90

STEAM GENERATOR FEEDRING J-NOZZLE REPLACEMENT

Summary:

Steam Generators are equipped with a feedring and J-nozzle to distribute the incoming feedwater within the steam generator. The function of the J-nozzles is to direct water in the downward direction, thus mitigating the potential for water hammer in the event the feedring were to become uncovered when the water level in the steam generator is lowered.

Some years ago, wall-thinning was observed in the carbon-steel J-nozzles which form part of the feedring in recirculating steam generators. The causative mechanism was attributed to erosion-corrosion of the carbon steel J-nozzles. Laboratory testing programs were conducted to study environmental factors relevant to the corrosion-erosion mechanism. J-nozzle wall thickness inspection data from operating steam generators were obtained and an improved J-nozzle design, featuring erosion/corrosion resistant materials, was developed to avoid this type of degradation.

SAFETY EVALUATION

The steam generator feedring J-nozzles do not perform a Safety Related function, are not required to maintain the reactor coolant system pressure boundary, and do not adversely impact plant safety or operation. However, the feedring is seismically supported to preclude potential interactions with Safety Related equipment. The modification associated with this EP has no adverse effect on safety related components or systems, and does not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval to implement this change was not required.

PLANT CHANGE/MODIFICATION 89-183

PC/M CLASSIFICATION	:	SR
UNIT	:	3 & 4
TURN OVER DATE	:	03/12/90

AFW TRIP & THROTTLE VALVE CONTROL CIRCUIT MODIFICATION

Summary:

In an attempt to manually open the Trip and Throttle (T & T) valve MOV 6459A from the Control Room, the valve motor was tripped by its thermal overload elements. The trip was caused by holding the manual control switch in the "OPEN" position for a longer time than necessary. The modification was necessary to provide an interlock in the manual control circuit to prevent such a trip.

This Engineering Package provided modification to the T & T valves MOV-6459A, B, C control circuits to prevent motor trip on thermal overload during manual operation of these valves.

The modification will protect the valve motor from tripping due to thermal overload caused by unnecessary prolonged control switch operation. This modification will enhance the control of the T&T valves without affecting the existing functional operation.

The T&T valves and the associated controls are Nuclear Safety Related and are part of the Auxiliary Feedwater system, which is required for safe shutdown when Normal Feedwater supply is not available. The modification per this PC/M involve internal wiring changes in Local Control Panel C238, C239 and C240 which are Safety Related, and accordingly, this PC/M is classified as Safety Related.

The modification did not add any equipment or components that could create any new safety hazards of any nature and will not affect the plant safety or operation.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 89-307

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	01/24/90

MAIN FEEDWATER SOLENOID VALVE REPLACEMENT

Summary:

This Engineering Package provided the design for replacement of the Solenoid Valves which provide the safety related closure of the Main Feedwater Valves. These Solenoid Valves are required to vent the instrument air from the Main Feedwater Valve actuator allowing the actuator spring to close the valve. These Solenoid Valves are normally energized and are de-energized to close the Main Feedwater Valves. SV-3-478A&C, SV-3-488A&C and SV-3-498A&C are redundant Solenoid Valves operated from train A & B respectfully and provide closure of their respective Main Feedwater Valve upon receipt of a safety injection signal or high steam generator level (80%) and must accomplish closure within 6.8 seconds or less. SV-3-478B, SV-3-488B and SV-3-498B are required to close their respective Main Feedwater Valve in 20 seconds or less upon receipt of a reactor trip signal coincident with TAVG less than 554 degrees F.

These modifications will improve the closure times of the subject FCV's and provide better spare parts availability for the solenoid valves.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 89-329

PC/M CLASSIFICATION	:	SR
UNIT	:	3 & 4
TURN OVER DATE	:	03/06/90

INSTALLATION OF ISOLATION VALVES FOR AND REPLACEMENT OF PI-205 A & B

Summary:

This Engineering Package covered the design and installation of Isolation Valves for, and replacement of PI-205 A & B, Air to Pinion Engaging Air Motor, pressure indicators. The replacement pressure gauges and the new isolation valves are part of the Emergency Diesel Generator (EDG) A & B air start systems. Their proper operation is necessary for the EDGs to perform their design safety function. This modification was made in order to enhance the performance of the system, and to improve the personnel and equipment safety during operations. Specifically, the existing pressure gauges, which indicate EDG A & B air start system pressure, were obsolete and required replacement. The Isolation Valves were installed so that the gauge can be isolated without disabling the entire air start system.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 89-352

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	09/14/89

ADDITION OF ANTI-SWEAT INSULATION ON CCW PIPING

Summary:

This Engineering Package provided the necessary design, documentation, and instructions to install anti-sweat insulation on portions of the Component Cooling Water (CCW) system piping located directly above the seal table. This insulation will prevent condensation from accumulating on the CCW piping and subsequently dripping on the seal table.

1-1/2 inches of "Foamglas" insulation was installed on a portion of the CCW piping to and from the "B" Control Rod Drive Mechanism coolers. The piping is located directly above the RCS seal table. The insulation chosen is appropriate for the application, its weight does not adversely impact existing stress, support analyses, or affected structures and it is encapsulated to prevent the insulation from reaching the sump as a result of LOCA induced forces. Also, the insulation meets the criteria of Reg. Guide 1.36 for material compatibility.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 89-375

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	03/20/90

BONNET EQUALIZATION LINES FOR MOV-3-750 AND MOV-3-751

Summary:

During the performance of Alternate Shutdown Testing, it was discovered that MOV-4-751 would not open in response to the Control Room switch. Opening of the valve by manual operation was successfully performed with followup electrical testing indicating that there was no problem within the motor operator. Further investigation and analysis indicated that the failure to open was likely to be due to hydraulic locking of the valve. This phenomenon occurs when the RCS pressure is lowered trapping water at a high pressure inside the bonnet and between the discs and seats. Therefore, the pressure in the bonnet must be equalized with line pressure. MOV 3-750 and MOV-3-751 serve as RHR pump suction valves with identical design configurations to MOV-4-751. Therefore, it is anticipated that MOV-3-750 and MOV-3-751 could be susceptible to the same hydraulic lock.

This Engineering Package removed the lower set of valve packing and installed bonnet equalizing lines from the valve packing leak-off connection of both valves MOV-3-750 and MOV 3-751 to the RCS side of the valve. For MOV-3-750, this source is the RCS bypass loop manifold instrument header at drain valve 3-562C. For MOV-3-751, this is the instrument line at valve 3-750C. Therefore, the bonnet will be at the same or lower pressure than that associated with the high pressure side.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 89-529

PC/M CLASSIFICATION	:	SR
UNIT	:	3 & 4
TURN OVER DATE	:	05/07/90

REVISION OF EQ DOCUMENTATION PACKAGE FOR MOV 3/4-866A & B

Summary:

This Engineering Package (EP) provided the mechanism for revising Environmental Qualifications Documentation Packages (EQ Doc Pac) 17.0 for limitorque Valve Actuators and 1001 for the Generic Approach and Treatment of Issues. These revisions were required to change the qualified post accident operating time of SI to RCS hot leg isolation valves MOV 3/4-866A & B from 2 hours to 31 days in accordance with vendor qualification documentation and engineering analysis. The 31 day qualified post accident operating time is consistent with the valves' actual qualification. The change in operating time for these valves requires that the radiation dose rate on the EQ Doc Pac System Component Evaluation Worksheets (SCEWs) be revised to reflect a 31 day post-accident dose in lieu of the 2 hour dose.

In addition, a document discrepancy regarding the EQ Doc Pac SCEWs radiation doses was noted for valves MOV 3/4-750 and 751. These SCEWS were revised to be consistent with the 31 day post-accident radiation dose.

There were no physical component/system changes associated with this Engineering Package.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP does not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 90-042

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	06/14/90

DEBRIS RESISTANT FUEL ASSEMBLY DESIGN - REGION 14

Summary:

This Engineering Package (EP) addressed a design modification to the Turkey Point 3 Region 14 fuel that implements a debris resistant fuel assembly design change beginning in Cycle 12. The design change consisted of the lengthening of the lower end plugs of the fuel (IFBAS[Internal Fuel Burnable Absorber] and non-IFBA) rods from 0.497 to 1.878 inches and the lowering of the spacer grids 1 through 6 by 1 inch. The basic concept of this feature is to displace the active portion of the fuel axially upward. Any debris that would be caught by the lowermost Inconel grid will now interact with the inert lower plug instead of the fuel rod cladding. The Debris Resistant Fuel Assembly design is a slightly modified version of the Optimized Fuel Assembly which currently exists in the core.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

PLANT CHANGE/MODIFICATION 90-247

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	06/07/90

REPLACEMENT OF REACTOR VESSEL O-RING SEALS

Summary:

During heat up of the Unit 3 reactor, a high temperature alarm signal was detected in the Reactor Vessel Flange Leak Detection System originating at the inner, metallic O-ring seal. This O-ring is one of two that provide the seal between the reactor head and the vessel. Although it is acceptable to operate with the inner seal leakage, Unit 3 was shutdown to investigate the source of the leakage.

Data was collected which confirmed that the high temperature alarm correctly signaled a leak past the inner O-ring. During these initial investigations no leakage was discovered through the outer O-ring.

As a potential corrective action, a change in the standard O-ring design is being evaluated. The O-rings currently in use at Turkey Point are silver plated, 304 stainless steel. The replacement O-rings under consideration are silver plated Inconel 718. This PC/M was issued to provide the engineering and safety evaluations necessary for installation of the Inconel 718 O-ring which provides enhanced sealing capability due to the increased radial (sealing) force afforded by the stronger inconel material as well as improved recovery following compression.

SAFETY EVALUATION

The modifications, as demonstrated in the Safety Evaluation do not constitute an unreviewed safety question nor require any changes to plant Technical Specifications. Therefore, this EP did not require NRC approval prior to implementation.

SECTION 2

Changes to the facility or procedures as described in the Safety Analysis Report not performed by a PC/M, and tests or experiments not described in the Safety Analysis Report.

SAFETY EVALUATION: JPN-PTN-SECJ-89-072 Revision 1

FIRST LOADING OF OVERHANG ROWS IN THE SPENT FUEL STORAGE RACKS

On November 21, 1984, the NRC approved License Amendments 111 and 105 for the expansion of the spent fuel pool capacity for Unit 3 and 4. The amendment was based on a Westinghouse analysis which concluded that lift off from the pool liner would not occur during a seismic event. Westinghouse subsequently informed FPL that if an "overhang row first" loading pattern were employed, lift off could occur. Westinghouse concluded, however, that the acceptance criteria of the NRC's "OT" position for Review and Acceptance of Spent Fuel Storage and Handling Applications in relation to tilting would still be met even if lift off did occur. As a result of the lift off concern, administrative controls were implemented by Change Request No 3 to PC/M 84-47 to ensure proper fuel placement to avoid rack lift off. During the review of the Westinghouse reanalysis, the NRC notified FPL that their review would only be necessary if the change involved a Technical Specification amendment or an unreviewed safety question. This safety evaluation demonstrated that the first loading of overhang rows in the spent fuel racks does not constitute a change to the basis supporting the license amendments and does not involve an unreviewed safety question or involve a change to the Technical Specifications. There is no physical change to the plant. There is no affect on plant operation and safety. There are no changes in design or operating practices or philosophy and there are no restrictions on plant operation. Therefore, prior NRC approval was not required for first loading of overhang rows in the spent fuel racks and consequently the administrative control, established to preclude rack lift off, that is currently in place on placement of spent fuel could be deleted.

Revision 1 incorporates the Plant Nuclear Safety Committee comments to explicitly address the acceptability of first loading of overhang rows on the storage of spent fuel. A draft of this safety evaluation was issued in 1986. The draft was revised to reflect the current status and includes necessary updated forms. The technical contents are adequate to conclude that tho overhang row first fuel loading and consequent removal of the administrative control to preclude rack lift off do not involve an unreviewed safety question or a change to the Technical Specifications.

Safety Evaluation Summary

The deletion of the administrative controls, currently in place restricting overhang row first loading, does not involve an unreviewed safety question or involve a change to the Technical Specifications.

Issued: November 11, 1989

This page was intentionally left blank.

SAFETY EVALUATION: JPN-PTN-SECJ-89-092 Revision 0

PCM 88-365 AND 88-363 SECURITY UPGRADE PROJECT DUCTBANK SOIL BORING SAFETY EVALUATION

The Security System upgrade at Turkey Point Units 3 and 4 under PC/M. 88-365 and 88-363, required a subsurface investigation involving soil borings and testing to determine radiological contamination levels of soil at or nearby the proposed ductbank and one lighting stanchion location. All work was performed in accordance with current industry standards.

Safety Evaluation Summary

The activities required to perform the soil investigation for the ductbank and light stanchion did not impact the function or integrity of any equipment, components, or structures that are required for the safe operation of the plant. Furthermore, this testing did not give rise to any unreviewed safety questions.

Issued: September 18, 1989

SAFETY EVALUATION: JPN-PTN-SECS-89-110 Revision 0

TEMPORARY LEAD SHIELDING FOR UNIT 3 UPPER REACTOR CAVITY

Temporary Lead Shielding was installed inside Unit 3 containment upper cavity to reduce the dose rates generated by the CRDM coils and components. The Temporary Lead Shielding was installed in modes 5 and 6 (when the reactor head is on), depressurized and was removed prior to RCS pressurization and entering Mode 4.

The evaluation shows that installation of Temporary Lead Shielding does not affect any Technical Specification, plant operation or safety.

Safety Evaluation Summary

Since no existing safety analyses are affected and no new failure modes are introduced, the installation of temporary shielding at this location does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR 50.59. Therefore, prior NRC approval was not required.

Issued: October 25, 1989

SAFETY EVALUATION: JPN-PTN-SECS-89-120 Revision 0

TEMPORARY PLACEMENT OF THE BATTERY DISCHARGE TESTER IN THE DC EQUIPMENT ROOM.

This Safety Evaluation analyzed the seismic response of a Propel Inc., Model HRD-1400-141, 3-Step, Hi-rate Discharge Tester which is being stored in the DC Equipment Room. The response of the discharge tester to both a postulated Operating Basis Earthquake and a Safe Shutdown Earthquake was analyzed.

The results of the analysis show that the discharge tester would rock but not overturn. It may be temporarily stored in the DC Equipment Room if all four wheels are locked. In addition, a clearance distance at its base of 1 inch and at its top of 6 inches must be maintained between the discharge tester and adjacent structures, systems and components.

Since this evaluation addressed structures, components and systems which are required to remain functional during an OBE and SSE, this evaluation is classified as Quality related. However, the discharge tester is required for maintenance testing only and is not considered Quality Related.

Safety Evaluation Summary

Based on the discussions contained in the safety evaluation, the temporary storage of the battery discharge tester within the DC Equipment Room did not result in an unreviewed safety question or reduce the margin of safety as define in the Technical Specifications.

Issued: November 2, 1989

SAFETY EVALUATION: JPN-PTN-SECS-89-122 Revision 2

TEMPORARY SHIELDING REQUEST NO. 90-02 AND 90-18

Temporary lead shielding was installed inside Unit 3 Containment at Elevation 14'-0" between the regenerative heat exchanger and LCV-3-460 valve and for the chemical and volume control system letdown piping upstream and downstream of the valve. The shielding was required to reduce the dose rates generated by the regenerative heat exchanger, LCV-3-460 valve and the letdown piping while working on Valve LCV-3-460. The temporary shielding was installed while Unit 3 was defueled and was removed prior to entering mode 6.

This safety evaluation concludes that the installation of the temporary lead shielding has no adverse impact on plant safety or operation and does not affect any Technical Specification.

Safety Evaluation Summary

Since no existing safety analysis is affected and no new failure modes are introduced, this work does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR50.59.

This evaluation demonstrates that the subject activity does not involve an unreviewed safety question or change to the Technical Specification pursuant to 10CFR 50.59, and, therefore prior NRC approval for this activity was not required.

Issued: February 4, 1990

SAFETY EVALUATION: JPN-PTN-SECS-89-124 Revision 0

TEMPORARY SHIELDING REQUEST NO. 90-03

Temporary lead shielding was installed inside Unit 3 Containment at El. 14'-0" - outside the bioshield wall, at the reactor cavity drain valve area. The shielding was required to prevent contact with hot spots under the drain valves and associated pipes and to reduce the general area dose rate.

The temporary shielding was installed while Unit 3 was in Mode 5 or defueled with the refueling cavity drained empty.

Safety Evaluation Summary

There are no Technical Specifications Restrictions in the Current, Interim, or Revised Technical Specifications affected by implementation of the temporary lead shielding because the installation of the shielding does not adversely interact with any safety related structure, systems, or components in the modes in which it is installed. The evaluation demonstrates that the subject activity does not involve an unreviewed safety question or change to the technical specification pursuant to 10CFR50.59, and prior NRC approval for this activity was not required.

Issued: December 18, 1989

SAFETY EVALUATION: JPN-PTN-SECS-89-127 Revision 0

RD-11 SHIELD BLOCK MOUNTING CONFIGURATION FOR POST ACCIDENT MONITORING

Process Radiation Monitor System (PRMS) Channel RD-11 was taken out of service for maintenance. During the course of that activity, two of the four bolts which secure the integral lead shielding for the RD-11 Detector were broken due to seizure within the connection threaded insert. The Safety Evaluation evaluated the acceptability of a return to service of RD-11 without the use of two bolts. PRMS Channel RD-11 monitors containment airborne particulate activity and is part of a sampling skid with RD-12. RD-12 monitors containment gaseous activity. Each of these monitors indicates and records containment atmospheric activity, alarms at a given setpoint, and initiates containment and control room ventilation isolation.

Safety Evaluation Summary

Since this evaluation demonstrates the operability of RD-11 in the condition described above, plant operation and nuclear safety are not adversely impacted.

This evaluation demonstrates that RD-11 will perform its design function and not adversely affect any surrounding equipment. Therefore, the use of RD-11 with only two bolts securing the internal lead shield does not constitute an unreviewed safety question or a change to any technical specification pursuant to 10CFR 50.59.

Technical Specifications 3.1.3 and 3.9.2 require RD-11 to be operable. There are no Technical Specifications affected since the removal of two of four mounting bolts does not affect the operability of RD-11 as discussed in the safety evaluation. The evaluation as contained here demonstrates that operations of RD-11 with only two of the four mounting bolts for the internal lead shield does not involve an unreviewed safety question or change to the Technical Specification pursuant to 10CFR 50.59, and prior NRC approval for this activity was not required.

Issued: November 24, 1989

SAFETY EVALUATION: JPN-PTN-SECS-89-133 Revision 0

TEMPORARY LEAD SHIELDING REQUEST NO. 90-09

Temporary Lead Shielding was installed inside Unit 3 Containment, at Elevation 14'-0" - inside the bioshield wall, around a portion of the RHR line between valve MOV-3-750 and loop C hot leg (up to Elevation 20'). The shielding was required to protect maintenance personnel from excessive radiation exposure generated by RHR piping. The lead shielding, supported from a free standing support, was in place only while the unit was defueled and was removed prior to entering Mode 6.

Safety Evaluation Summary

This safety evaluation concludes that the installation of the temporary lead shielding has no adverse impact on plant safety or operation and does not affect any Technical Specification.

No existing safety analysis is affected and no new failure modes are introduced as determined in the safety evaluation, this work does not constitute an unreviewed safety question or require a change to any Technical Specifications pursuant to 10CFR50.59. Therefore, prior NRC approval was not required for implementation of the temporary shielding request.

Issued: January 11, 1990

SAFETY EVALUATION:JPN-PTN-SECS-89-137 Revision 0

TEMPORARY SHIELDING REQUEST NO. 90-03

Temporary Lead Shielding was installed inside Unit 3 Containment at Elevation 14'-0" for the RHR piping outside the Bioshield Wall, specifically, the 14" return line between MOV-3-751 and the Bioshield Wall. The shielding was required to reduce personnel radiation exposure while performing maintenance on adjacent equipment.

The temporary shielding was installed while Unit 3 was defueled and was removed prior to entering Mode 6.

Safety Evaluation Summary

Since no existing safety analysis is affected and no new failure modes are introduced as demonstrated in the Safety Evaluation, this work does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR50.59. Therefore, prior NRC approval was not required for implementation of the temporary shielding request.

Issued: December 22, 1989

SAFETY EVALUATION: JPN-PTN-SECS-90-002 Revision 0

TEMPORARY SHIELDING REQUEST NO. 90-04

Temporary Lead Shielding was installed inside Unit 3 Containment, outside the bioshield wall, at Elevation 14'-0" to protect maintenance personnel from receiving excessive radiation exposure generated by the piping adjacent to charging valves CV-3-310A, CV-3-310B and CV-3-311 (including the valve bodies). The lead shielding, supported from a free standing support, was in place only while the unit was defueled and was removed prior to entering Mode 6.

Safety Evaluation Summary

This safety evaluation concludes that the installation of the temporary lead shielding has no adverse impact on plant safety or operation and does not affect any Technical Specification.

No existing safety analysis is affected and no new failure modes are introduced as determined in the safety evaluation, this work does not constitute an unreviewed safety question or require a change to any Technical Specification pursuant to 10CFR50.59. Therefore, prior NRC approval for this activity was not required.

Issued: January 16, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-018 Revision 0

UNIT 3 SPENT FUEL POOL KEYWAY GATE BOOT SEAL REPLACEMENT

The inflatable seal on the Unit 3 Spent Fuel Pool (SFP) Keyway Gate was replaced to provide an adequate seal between the SFP and the transfer canal. The replacement of the keyway seal did not require a safety evaluation, pursuant to 10 CFR 50.59, since it is a direct one-for-one replacement. However, the removal and reinstallation of the Keyway Gate is necessary for the replacement of the seal. This procedure constitutes a heavy load lift and requires a safety evaluation to assure compliance with the requirements of NUREG-0612.

Safety Evaluation Summary

This evaluation demonstrates that the removal and reinstallation of the Keyway Gate does not affect plant operation or safety, or any Technical Specifications.

Since no existing safety analyses are affected and no new failure modes are introduced, the removal and reinstallation of the Keyway Gate does not constitute an unreviewed safety question or a change to any Technical Specifications pursuant to 10 CFR 50.59. Therefore, prior NRC approval was not required.

Issued: January 25, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-019 Revision 0

TEMPORARY LEAD SHIELDING REQUEST NO. 90-19

Temporary Lead Shielding was installed inside Unit 3 containment to the upper platform of Reactor Vessel Head Lifting Rig. The purpose of the Temporary Lead Shielding was to protect maintenance personnel from excessive radiation exposure generated by the CRDM coil and components.

Safety Evaluation Summary

This safety evaluation concluded that the installation of the temporary lead shielding in accordance with Attachment No. 1 has no adverse impact on plant safety or operation and does not affect any Technical Specification.

Since no existing safety analysis is affected and no new failure modes are introduced, this work does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR50.59.

Issued: January 31, 1990

SAFETY EVALUATION: JPN-PTN-SEEJ-89-075 Revision 1

PLANT CONDITIONS EXISTING PRIOR TO IMPLEMENTATION OF PC/M 89-420

In the 1983 Fire Protection Program Report (FPPR) and subsequent supplemental FPRR information, commitments were made for performing certain fire protection modifications. For the Charging Pump Rooms, a commitment was made to protect the local control stations for Charging Pumps 3B and 4B by providing an enclosure fabricated from 1-hour rated materials, or to relocate it out of the fire zone, and to provide the motor power cables for the "A" and "B" charging pumps with 1-hour rated fire protection. For the Feedwater Platform fire zones commitments were made to reroute the cables for AFW flow control valves to maintain a minimum of 20 feet of separation from the cables of the corresponding redundant valves and to provide 1-hour fire rated protection where this minimum separation could not be achieved.

Subsequent to the issuance of the FPRR, schedular exemptions from the requirements of Appendix R were granted to Turkey Point Units 3 and 4. The schedular commitments for cable rerouting and raceway protection modifications necessary to achieve compliance with Appendix R were deferred to June 1986 and September 1986, respectively. In conjunction with these schedular exemptions, compensatory measures in the form of roving fire watches were established in the fire areas for which schedular exemptions were requested.

Engineering Package (EP) for PC/M 89-420 provided for the installation of fire rated assemblies (raceway protection) on electrical raceways and supports which are required to prevent the included cables from being damaged by an Appendix R postulated fire. The raceways to be protected are associated with the Charging Pump "A" Local Control Stations and pump motor power cables located in Fire Zones 45 (Unit 4 Charging Pump Room) and 55 (Unit 3 Charging Pump Room). The cables associated with the local control station will be provided with 3-hour fire rated material and the pump motor power cables, which are routed in flexible steel conduit, will be provided with 1-hour fire rated material. Also, the conduits containing the control cables for solenoid valves SV-*-2914, -2916, and -2918 which are located in Fire Zones 113 (Unit 4 Feedwater Platform -El. 18.00', 38.50') and 116 (Unit 3 Feedwater Platform - El. 18.00', 39.75') will be provided with 1-hour fire rated material. These solenoid valves are associated with the Auxiliary Feedwater (AFW) Flow Control Valves CV-*-2816, -2817, and -2818.

The purpose of this safety evaluation was to demonstrate that, in the unlikely event of a fire occurring in any of the above zones, the interim plant condition that existed from September 1986 and prior to implementation of the modification to the above raceways

provided in the EP for PC/M 89-420 was acceptable with respect to plant safety and operation. This plant condition was such that the levels of fire protection provided and the implementation of compensatory measures were consistent with the existing fire hazards for the fire zones evaluated. The EP enhanced the existing level of fixed fire protection in the fire zones evaluated.

This evaluation justifies the plant condition with respect to fire protection features and associated plant safety and operation existing prior to the implementation of fire protection modifications provided by the EP for PC/M 89-420. Since this evaluation addresses areas containing safety related equipment for which commitments associated with fire protection have not been completed, the evaluation is classified as Safety Related.

Based on the discussions contained in this evaluation, the plant condition existing prior to implementation of the modification to the above raceways provided in the EP for PC/M 89-420 is acceptable with respect to plant safety and operation since the existing levels of fire protection features and the compensatory measures (roving fire watches) provided for the Charging Pump Rooms and the Feedwater Platforms were consistent with the fire hazards in these zones and also provided a high level of assurance that at least one train of safe shutdown cables and equipment would have remained free of fire damage.

Safety Evaluation Summary

Based on the discussions contained in this evaluation, the plant condition existing prior to implementation of the modifications to the raceways provided in PC/M 89-420 is acceptable with respect to plant safety and operation since the existing levels of fire protection features provided for the Charging Pump Rooms and Feedwater Platforms were consistent with the fire hazards in these areas. The implementation of conservative compensatory measures in the form of, at least hourly, roving fire watch patrols to monitor the 18 foot elevations containing safety related cables and the Auxiliary Building fire areas in addition to the fixed and portable fire protection features available for use provides a high level of assurance that at least one train of safe shutdown cables and equipment will remain free of fire damage.

Issued: September 27, 1989

SAFETY EVALUATION: JPN-PTN-SEEJ-89-076 Revision 1

PLANT CONDITIONS EXISTING PRIOR TO REMOVAL OF THE RCGVS VALVE FUSES

In order to address the requirements for Hi-Lo pressure interface relative to a spurious actuation of the Reactor Coolant Gas Vent System (RCGVS) valves (SV-*-6318A&B, -6319A&B, -6320A&B), the motive power fuses were described, in the Turkey Point Units 3 and 4 UFSAR and in the Fire Protection Program Report of June 1983, to be administratively removed during power operation. These fuses have not been removed during power operation, as described. Guidelines for revising existing plant procedures have been provided to the Plant by Engineering to ensure that RCGVS valves remain closed during a fire by administratively removing the motive power fuses.

Subsequent to the issuance of the Fire Protection Program Report, schedular exemptions from the requirements of Appendix R were granted to Turkey Point Units 3 and 4. The schedular commitments for completion of modifications for compliance with Appendix R safe shutdown capability were deferred to September 1986.

The purpose of this safety evaluation is to justify and document the acceptability of operation of Turkey Point Units 3 and 4 prior to the procedure changes which require removal of the motive power fuses.

Safety Evaluation Summary

This safety evaluation involves plant systems and equipment which are necessary to maintain the integrity of the Reactor Coolant System (RCS) pressure boundary. Therefore, this safety evaluation is classified as Safety Related.

Based on the discussions contained in this evaluation, the plant condition existing prior to removal of the motive power fuses for the RCGVS valves is acceptable with respect to plant safety and operation. This conclusion is justifiable since the most severe consequences associated with the failure to administratively remove the motive power fuses for the RCGVS would have been a breach in the RCS pressure boundary which, by the design of the RCGVS, is well within the makeup capacity of a single charging pump. However, the motive power fuses for the RCGVS valves were removed prior to implementation and/or approval of this safety evaluation.

Based on the discussions contained in this evaluation, the failure to remove the motive power fuses associated with the RCGVS valves, to prevent spurious operation in the unlikely event of a fire, is acceptable with respect to plant safety and operation, and did not result in an unreviewed safety question or a reduction in the

margin of safety as defined in the bases for any Technical Specifications. In accordance with the guidance contained in NRC Generic Letter 86-10, Section 5.3.10, this evaluation demonstrates that the safe shutdown capability is not adversely affected by the spurious opening of the redundant isolation valves in any one of the high-low pressure interface Reactor Coolant Gas Vent System lines.

Issued: November 7, 1989

SAFETY EVALUATION: JPN-PTN-SEEJ-89-104 Revision 0

PLANT CONDITION EXISTING PRIOR TO INSTALLATION OF AN ISOLATION SWITCH FOR THE CVCS LETDOWN ISOLATION VALVES

Consistent with the guidance provided in Generic Letter 86-10, the purpose of this Safety Evaluation was to demonstrate that the safe shutdown capability of Turkey Point Units 3 and 4 is not adversely affected by the spurious opening of the redundant isolation valves in any one reactor coolant system letdown path, which is a high-low pressure interface. This safety evaluation involves plant systems and equipment which are necessary to maintain the integrity of the RCS pressure boundary. Therefore, this safety evaluation is classified as Safety Related.

Letdown must be isolated during a postulated Appendix R fire to achieve hot shutdown to ensure that the Reactor Coolant System high-low pressure interface is not breached. Circuit interlocks from all three CV-*-200 valves in the LCV-*-460 control circuit prevent closing of LCV-*-460 if any of the CV-*-200 valves are open. This interlock is intended to prevent potential damage to the regenerative heat exchanger due to pressure surges caused by closure of LCV-*-460 prior to closure of all the CV-*-200 valves and subsequent repressurization of the line upon opening valve LCV-*-460.

For fire scenarios where credit is taken to close LCV-*-460, spurious opening actuation of any of the CV-*-200 valves can potentially disable the closure of LCV-*-460. The permanent modification is to install switches in the Control Room which can defeat the interlock. Until the modification is implemented, the situation can be mitigated through operator actions, such as pulling fuses in the Control Room to defeat the interlock, locally operating handwheels on isolation valves or isolating instrument air to effect valve closure. In the interim, however, specific instructions for defeating the interlock have been provided to pull the fuses.

Although pulling fuses for safe shutdown equipment is not specifically prohibited by Appendix R requirements, it is inconsistent with NRC guidance for actions to achieve hot shutdown. From a manual actions standpoint, however, this provision is prudent because the required actions can be accomplished quickly and without leaving the Control Room. This provision will serve until permanent modifications are implemented to provide an isolation switch inside the Control Room to defeat the circuit interlock.

Safety Evaluation Summary

Based on the discussions contained in this Safety Evaluation, the plant conditions existing prior to implementation of the proposed isolation switch modification are acceptable with respect to plant safety and operation, do not involve an unreviewed safety question and do not require any changes to Technical Specifications.

Issued: November 9, 1989

SAFETY EVALUATION: JPN-PTN-SEEJ-90-039 Revision 0

UNIT 3 ENGINEERING SAFEGUARDS INTEGRATED TESTING

Integrated Safeguards Testing (IST) is performed on Unit 3 at each refueling interval to meet the commitment for demonstrating the readiness of the onsite emergency power system and vital equipment to achieve and maintain a safe shutdown condition in one unit while mitigating the postulated design basis accident on the other unit, assuming loss of offsite power to both units. The Unit 3 IST is presently conducted in accordance with procedure 3-OSP-203 on both 4160 volt safety related busses (trains) simultaneously while Unit 3 is at Cold Shutdown and Unit 4 is in Hot Shutdown. Procedure 3-OSP-203 is being split into 3-OSP-203.1 for Train A and 3-OSP-203.2 for Train B to conduct the Unit 3 IST on an individual bug (train-by-train) basis while Unit 3 is at cold shutdown and Unit 4 is in Hot Standby (Mode 3) or any less restrictive mode of operation.

Turkey Point has committed to the NRC to perform IST on a train-by-train basis to assure proper function of equipment powered by Telemand Transferred Motor Control Centers.

The method of conducting the revised Unit 3 IST is bounded by the Current, Interim, and Revised Technical Specifications. The onsite emergency power system will be available to power the minimum Engineered Safety Feature (ESF) loads on any unit and the loss of offsite power loads on both units, at all times during the Unit 3 IST. A review of equipment shared between units, including the Emergency Power System and the Engineered Safety Features, has been performed to ensure that Unit 4 is maintained in Hot Standby and the availability of equipment required for safe shutdown or accident mitigation of Unit 4 is not reduced beyond Technical Specification limitations.

All the Technical Specification surveillance requirements presently satisfied by the Unit 3 IST will continue to be satisfied when the procedure is performed on an independent train basis. Testing by individual trains allowing the non-test EDG to be available throughout the course of the IST to respond to an actual LOOP event, and the surveillance of equipment which received input from both safeguards and/or emergency power trains will be enhanced since the response to the actuation signals of both trains will be tested independently.

Safety Evaluation Summary

This safety evaluation addresses the technical and licensing requirements for the proposed Unit 3 IST procedure revisions discussed above and concludes that an unreviewed safety question does not exist and a change to the Plant Technical Specifications is not required. The basis for this conclusion consists of the independent train testing methods which ensure that the onsite emergency power system and all vital safety related equipment are not reduced below Technical Specification limitations, and that they will be available and operable for Unit 4 during the Unit 3 IST. The basis for this conclusion also includes compliance with certain requirements and restrictions to complement these testing methods as specified in this safety evaluation.

This safety evaluation has determined that the effects of the proposed revised Unit 3 IST methodology will remain within the bounds of the plant licensing bases and safety limits that have previously been reviewed and approved by the NRC. Therefore, prior NRC approval is not required to implement the proposed revised Unit 3 Integrated Safeguards Testing on a train-by-train basis.

Issued: April 2, 1990

SAFETY EVALUATION: JPN-PTN-SEES-89-062 Revision 0

RESTORATION OF R-18 PRMS CHANNEL TO SERVICE USING THE R-3-17B PRMS CHANNEL CABLE INSTALLED BY TSA-3-90-67-48

A Temporary System Alteration (TSA) was requested to use an alternate cable to replace the damaged cable for the drawer of R-18, Waste Disposal System Liquid Effluent Monitor. The Turkey Point Updated FSAR identifies that the R-18 channel provides 1) remote indication on the Waste Disposal System control board, 2) control room indication, and 3) an interlock to operate RCV-018. The alternate cable is currently used on channel R-3-17B so implementation of this TSA will place R-3-17B out of service.

Safety Evaluation Summary

This TSA has no effect on plant operation and no safety concerns. This safety evaluation imposes no restrictions on plant operation. This TSA has no Unreviewed Safety Question.

Based on a review of Section 3.9 and 4.12 of the Current Technical Specifications Sections 3/4.3 and 3/4.11-1 of the Revised Technical Specifications and Plant Procedure 0-ADM-021, "Technical Specification Implementation Procedure" the implementation of TSA-3-90-67-48 does not require a change to the Technical Specifications.

Based on the evaluation, temporary rewiring of R-18 with cables from R-17B involves neither a change to the Technical Specification nor a unreviewed safety question. Therefore, prior NRC approval was not required to perform the associated TSA.

Issued: May 25, 1990.

SAFETY EVALUATION: JPN-PTN-SEES-90-035 Revision 0

TEMPORARY INSTALLATION OF A MOTOR STARTER / DISCONNECT FOR THE EMERGENCY SPENT FUEL POOL COOLING WATER PUMP (TSA NO. 3-90-33-18)

A motor starter / disconnect was temporarily installed inside Unit 3 Spent Fuel Pool Building near the emergency spent fuel pool (SFP) cooling water pump. The motor starter / disconnect mounted to a free standing support stand, may be in place during any mode of plant operation provided the requirements identified in the evaluation for the location of this equipment and its use for the operation of the emergency SFP cooling water pump are complied with.

Safety Evaluation Summary

This safety evaluation concludes that the temporary installation of the motor starter / disconnect mounted to a free standing support stand has no adverse impact on plant safety or operation and does not affect any Technical Specification.

No existing safety analysis is affected and no new failure modes are introduced as determined in the safety evaluation, therefore, the work does not constitute an unreviewed safety question or require a change to any Technical Specification pursuant to 10CFR50.59.

Issued: February 21, 1990

SAFETY EVALUATION: JPN-PTN-SEIJ-89-081 Revision 0

TEST FOR INSTRUMENTATION LOOP MODIFICATIONS

Nonconformance Report N-89-0709 identified instrumentation loops on several systems which have non-isolated inputs going to the Safety Parameter Display System (SPDS). These non-isolated inputs to SPDS have been determined to be causing excessive Loading on the primary instrument loop by a common mode voltage problem. The common mode voltage concerns were also identified by Technology for Energy Corporation. As a result of the SPDS tie-ins, these loops have been experiencing deviations in SPDS indicated values as well as in the primary loop instrumentation. To maintain the signals to the SPDS for these loops, a preliminary bench test was performed with the loop wiring being modified and the signal ground location changed. Based on this preliminary test a test procedure was prepared and submitted under a separate transmittal. This test will remove one of the loops from service in accordance with existing plant procedures and determine whether the modifications made will restore the accuracy of the loops involved back to acceptable limits. Additionally, a test will be made on the original loop with new integrated circuits on the SPDS input card to determine if this modification will restore the accuracy of the loop involved back to acceptable limits. The affected loops are listed below:

Loop No.	Description
TE-3-411 B; C	RCS Bypass Loop Temperature
TE-3-421 B; C	RCS Bypass Loop Temperature
TE-3-431 B; C	RCS Bypass Loop Temperature
FT-3-6277A; B; C	Steam Generator A/B/C Blowdown Flow
FT-3-6274	Blowdown Heat Exchanger to Condenser Flow

Reinstating the SPDS tie-in for each loop will have no effect on plant operation or safety, nor will it create any changes in operating practices since the test restores the loop to its previous configuration as outlined in PC/M 81-158 with only minor wiring changes. Since these loops will continue to provide their respective control and indication functions, there is no effect on plant operation.

Safety Evaluation Summary

The performance of these tests does not change the function or operational requirements of the existing loops or systems. There are no adverse effects on plant operation or plant safety and implementation of this test does not constitute an unreviewed safety question, nor require changes to the Plant Technical Specifications.

Issued: September 27, 1989

This page was intentionally left blank.

SAFETY EVALUATION: JPN-PTN-SEIJ-89-082 Revision 0

TEST OF INSTRUMENTATION LOOP MODIFICATIONS - U-4

Nonconformance Report N-89-0709 identified instrumentation loops on several systems which have non-isolated inputs going to the Safety Parameter Display System (SPDS). These non-isolated inputs to SPDS have been determined to be causing excessive loading on the primary instrument loop by a common mode voltage problem. The common mode voltage concerns were also identified by Technology for Energy Corporation (TEC) letter P-81-045A dated April 15, 1981. As a result of the SPDS tie-ins, these loops have been experiencing deviations in the SPDS indicated values as well as in the primary loop instrumentation.

To maintain the signals to the SPDS for these loops, a preliminary bench test was performed with the loop wiring being modified and the signal ground location changed. Based on this preliminary test a test procedure has been prepared and submitted under a separate transmittal. This test will remove one of the loops from service in accordance with existing plant procedures and determine whether the modifications made will restore the accuracy of the loops involved back to acceptable limits. Additionally, a test will be made on the original loop with new integrated circuits on the SPDS input card to determine if this modification will restore the accuracy of the loop involved back to acceptable limits. The affected loops are listed below:

Loop No.	Description
TE-4-411 B; C	RCS Bypass Loop Temperature
TE-4-421 B; C	RCS Bypass Loop Temperature
TE-4-431 B; C	RCS Bypass Loop Temperature
FT-4-6277A; B; C	Steam Generator A/B/C Blowdown Flow
FT-4-6274	Blowdown Heat Exchanger to Condenser Flow

Reinstating the SPDS tie-in for each loop will have no effect on plant operation or safety, nor will it create any changes in operating practices since the test restores the loop to its previous configuration as outlined in PC/M 81-158 with only minor wiring changes. Since these loops will continue to provide their respective control and indication functions, there is no effect on plant operation.

The performance of these tests does not change the function or operational requirements of the existing loops or systems. There are no adverse effects on plant operation or plant safety and implementation of this test does not constitute an unreviewed safety question nor require changes to the Plant Technical Specifications.

Safety Evaluation Summary

Reinstating the SPDS tie-in and modifying the loop wiring for each loop will have no effect on plant operation or safety, nor will it create any changes in operating practices since the test restores tho loop to its SPDS configuration. Since these loops will continue to provide their respective control and indication function, there is no effect on plant operation.

The performance of these tests does not constitute an unreviewed safety question nor require changes to the Plant Technical Specifications. Therefore, the performance of the test did not require prior NRC approval.

Issued: September 27, 1989

SAFETY EVALUATION: JPN-PTN-SEIJ-89-087 REVISION 1

G3 RPI RETURN TO SERVICE TEMPORARY SYSTEM ALTERATION 4-89-28-30

In accordance with revised Technical Specifications 3/4.1.3.5 and 3/4.1.3.1, during power operation only, one full length rod can be declared inoperable. Failure to obtain a rod position indication is then permitted for only one rod during power operation.

G3 RPI has an intermittent open circuit causing no signal to the rod position indicator (secondary coil open).

This TSA is to return G3 RPI to service modifying the signal conditioning module located in rack 4QR72 to accept the primary coil A.C. signal as input to the G3 RPI instead of the open secondary coil. This temporary modification is required to maintain the G3 RPI indication until the secondary coil can be repaired or replaced during the next cold shutdown outage. Revision one to this Safety Evaluation evaluates a change in the circuit to permit enhanced adjustability, as well as the full scale meter calibration. The results of a technical analysis performed by Westinghouse for this change are also incorporated.

Revision 0 of this document was prepared in support of TSA-4-89-28-29. That TSA was fully approved but not implemented. TSA 4-89-28-30, which Revision 1 to this safety evaluation supports, supersedes the previous TSA in its entirety. The conclusions of the previous evaluation are unaffected by this change.

Safety Evaluation Summary

This temporary modification does not change plant operation or safety and does not create an unreviewed safety question or change any Technical Specification.

Issued: September 18, 1989

SAFETY EVALUATION: JPN-PTN-SEIJ-89-125 Revision 0

OPERATION WITH THE MAIN STEAM SAMPLING VALVES OPEN FOR POST ACCIDENT MONITORING

Process Radiation Monitor RAD-6426 is the radiation monitor installed for post accident monitoring of the main steam lines. The monitor is located on the 30'-0" elevation next to the mezzanine sampling sink. The monitor receives a continuous steam blowdown through valves 3-107, 3-207 and 3-307 and valves 4-107A, 4-207 and 4-307. This monitor is independent of the steam generator blowdown radiation monitor that receives a sample from valves MOV-1425, MOV-1426, and MOV-1427. Currently, all valves required to keep RAD-6426 operable are kept open during normal operation. However, during a steam generator tube rupture, downstream valves 3-10 NNA-3A, 3B, 3C or 4-10-NNA-4A, 4B, 4C are required to be closed within 30 minutes of the S.G. tube rupture event.

This safety evaluation evaluates the effects of leaving the MS sample valves to RAD-6426 open during a steam generator tube rupture event and shows that such a change to plant procedures does not represent an unreviewed safety question pursuant to 10CFR50.59. The procedural change of allowing these valves to remain open during a tube rupture event will have no effect on plant operation or safety and will allow continued radiation monitoring of the steam lines as required by NUREG-0737. Implementation of this change does not constitute an unreviewed safety question nor require a change to the Plant Technical Specifications.

Safety Evaluation Summary

Leaving the main steam sample lines open during a steam generator tube rupture will have no effect on plant operation or safety, since the monitor will continue to provide its radiation monitoring function. The requirement for isolating the main steam sample valves associated with RAD 6426 during a steam generator tube rupture can be removed.

In conclusion, leaving the main steam sample valves open does not constitute an unreviewed safety question nor require changes to the Plant Technical Specifications. Therefore, leaving the main steam line sample lines open did not require prior NRC approval.

Issued: December 14, 1989

SAFETY EVALUATION: JPN-PTN-SEIJ-89-131 Revision 0

VENDOR TEST PROCEDURE MODULAR COMPUTER SYSTEMS CLASSIC III/95 INSTALLATION

This Safety Evaluation allows the temporary installation of the Modular Computer systems (MOD COMP) Classic III/95 for Unit 3.

Temporary unavailability of the Safety Parameter Display System (SPDS) during implementation of this test will have no adverse effect on plant operation or safety as the SPDS does not perform Nuclear-Safety related functions, and the brief disconnection will not affect primary safety-related or quality-related indications. Implementation of this test procedure did not constitute an unreviewed safety question nor does it require any changes to the Technical Specifications.

The test will evaluate the performance of Emergency Response Data Acquisition and Display System (ERDADS) in terms of speed of response and accuracy of the system as evidenced by such criteria as channel update speed.

The present ERDADS is too slow for its present functions. Therefore, FPL desires to increase the CPU speed and disk drive speed, by purchasing new hardware. Mod Comp has proposed the new Classic III/95 processors and T88XX disk drives. FPL has agreed to purchase this equipment if it significantly speeds up the ERDADS. Therefore, FPL plans to make a temporary system modification with the new CPU's and disk drives under temporary Mod Comp Test Procedure RTS 89-3762, while Reactor Engineering determines the speed response.

Safety Evaluation Summary

The current, proposed and interim Technical Specifications do not contain any references to the SPDS and, therefore, are not affected by this change. The implementation of RTS 89-3762 does not constitute an unreviewed safety question nor require changes to the Plant Technical Specification. Therefore, the modification did not require prior NRC approval.

Issued: December 7, 1989

SAFETY EVALUATION: JPN-PTN-SEIJ-89-132 Revision 0

VENDOR TEST PROCEDURE MOD COMP CLASSIC III/95 INSTALLATION

This Safety Evaluation a temporary installation of the Modular Computer systems (MOD COMP) Classic III/95 for Unit 4.

Temporary unavailability of the Safety Parameter Display System (SPDS) during implementation of this test will have no adverse effect on plant operation or safety as the SPDS does not perform Nuclear-Safety related functions, and the brief disconnection will not affect primary safety-related or quality-related indications. Implementation of this test procedure did not constitute an unreviewed safety question nor does it require any changes to the Technical Specifications.

The test will evaluate the performance of Emergency Response Data Acquisition and Display System (ERDADS) in terms of speed of response and accuracy of the system as evidenced by such criteria as channel update speed.

The present ERDADS is too slow for its present functions. Therefore, FPL desires to increase the CPU speed and disk drive speed, by purchasing new hardware. Mod Comp has proposed the new Classic III/95 processors and T88XX disk drives. FPL has agreed to purchase this equipment if it significantly speeds up the ERDADS. Therefore, FPL plans to make a temporary system modification with the new CPU's and disk drives under temporary Mod Comp Test Procedure RTS 89-3762, while Reactor Engineering determines the speed response.

Safety Evaluation Summary

The current, proposed and interim Technical Specifications do not contain any references to the SPDS and, therefore, are not affected by this change. The implementation of RTS 89-3762 does not constitute an unreviewed safety question nor require changes to the Plant Technical Specification. Therefore, the modification did not require prior NRC approval.

Issued: December 7, 1989

SAFETY EVALUATION: JPN-PTN-SEIS-89-029 Revision 0

CORE EXIT THERMOCOUPLES TEMPORARY CABLES FOR RCS TEMPERATURES MONITORING DURING MID-LOOP OPERATION

In order to monitor the reactor coolant temperature while the plant is in mid-loop condition (a mid-loop condition exists whenever RCS water level is below the top of the flow area of the hot legs at the junction with the reactor vessel), non-safety related temporary jumpers were installed between the reactor head connectors and the containment connectors, one for each channel, in place of the Mineral Insulated (MI) cables and prior to operating in a reduced inventory condition to assure continuous temperature indication of representative core-exit thermocouples.

This modification for Unit 3 is required as part of the expeditious action items recommended in Generic Letter 88-17 for the mid-loop operation because of the potential consequences of loss of shutdown cooling concurrent with significant core decay heat generation, during drain-down operations. Since the temporary cables are attached to the reactor head, a safety related component, the modification is classified as Quality Related.

Safety Evaluation Summary

This evaluation concludes that this modification does not adversely affect plant safety or operations, and that installing jumpers can be performed within limits of the Turkey Point Technical Specifications As discussed in this 10CFR50.59 evaluation, the addition of the jumpers does not constitute an unreviewed safety question or require a change to Technical Specifications. Therefore, prior NRC approval was not required to implement this change.

Issued: August 22, 1989

SAFETY EVALUATION: JPN-PTN-SEIS-89-103 Revision 0

REPLACEMENT OF SV-4-387 FOR TSA 4-89-47-33

The I&C Maintenance Department has requested a Temporary System Alteration (TSA) to install an ASCO model HT 831654, 125VDC for SV-4-387. The solenoid valve presently installed (ASCO model LBX 831614, 125VDC) is inoperable and is no longer manufactured.

This change has no effect on plant operation and is not a safety concern. This safety evaluation imposes no restrictions on plant operation. This change is not an unreviewed safety question nor does it require a change to the Technical Specifications.

SV-4-387 is a solenoid valve and is required for operation of the excess letdown heat exchanger isolation valve (CV-4-387). This solenoid valve does not perform any active safety function, however, it is required per Appendix R for shutdown of the plant from the alternate shutdown panel and therefore is classified Quality Related.

Safety Evaluation Summary

The safety evaluation demonstrates that the subject activity does not involve an unreviewed safety question or change to the Technical Specifications pursuant to 10CFR 50.59. Prior NRC approval for this activity was not required.

Issued: October 3, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-058 Revision 3

DOCUMENTATION CHANGES TO UFSAR APPENDIX 9.6A

As a result of the documentation review in preparation for the NRC Appendix R audit, discrepancies were noted in UFSAR Appendix 9.6A. These discrepancies were investigated to assure that there was no adverse impact to the plant. The investigation confirmed that the design configuration of the plant was correct and had not been affected.

This safety evaluation analyzes these discrepancies and documents the recommended changes to UFSAR Appendix 9.6A.

Revision 3 incorporates FPL comments and demonstrates that there are no effects on Current, Revised or Interim Technical Specifications.

Safety Evaluation Summary

Based upon the above 10CFR50.59 safety evaluation, the correction of discrepancies in UFSAR Appendix 9.6A will not result in an unreviewed safety question or require any changes to the current Technical Specifications. In addition, these UFSAR changes will not adversely affect plant operation or safety. Therefore, prior NRC approval was not required to implement these changes.

Issued: November 8, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-063 Revision 1

COMBUSTIBLE LOADING CHANGES TO UFSAR APPENDIX R FIRE HAZARD ANALYSIS

A review of the engineering calculation for the combustible loadings in each fire area revealed discrepancies between the engineering calculations and the information provided in the UFSAR, Appendix R, Fire Hazards Analysis (FHA).

Each Fire Zone described in the UFSAR was compared to the engineering calculation for combustible loadings to assess the potential impact on the UFSAR Fire Zone combustible loading tables. This comparison resulted in changes to the UFSAR combustible loadings for Fire Zones 10, 20, 25, 28, 41, 58, 59, 60, 61, 67, 93, and 94.

The FHA provided in UFSAR Appendix 9.6A was revised to include the accurate information from the engineering calculation. In the case of Fire Zone 58, the engineering calculation and UFSAR were revised following a plant walkdown. While the quantity of combustibles increased marginally for Fire Zone 58, the actual heat load decreased. A Fire Protection Engineer determined that there was no additional impact to the fire protection systems in each of the fire zones. This is only a documentation change to correct information in the UFSAR.

Safety Evaluation Summary

Based upon this evaluation, the proposed changes to the UFSAR are acceptable and will not result in an unreviewed safety question or result in any changes to the Plant Technical Specifications. In addition, these modifications to the UFSAR will not adversely affect plant operation and safety while ensuring compliance with Appendix R Safe Shutdown capability. Therefore, NRC approval prior to implementation of the UFSAR changes was not required.

Issued: November 3, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-089 Revision 0

FIRE WATER SUPPLY SYSTEM

In preparation for the 1989 NRC Appendix R audit, two concerns were expressed regarding Fire Water Supply System:

1. The fire water pumps, using Raw Water Storage Tanks I and II (RWTs I and II) cannot supply 300,000 gallons each of fire water at the maximum Fire Water Supply System demand of 1624 gpm (874 gpm for sprinkler or deluge system and 750 gpm for manual hose stream) because of the current fire pump suction design characteristics and vortex effects. This is not in accordance with the FPL commitment to the NRC to provide a redundant dedicated fire water supply of 300,000 gallons from RWTs I and II.

2. The Current Plant Technical Specifications, Section 3.14.2.a.(2), state that Fire Water Supply Systems shall be operable with separate water supplies, with a minimum of 30,000 gallons in the elevated storage tank and 150,000 gallons in the Raw Water Tank (RWT). The Current Plant Technical Specifications do not reflect the current plant condition for the redundant 300,000 gallon fire water supplies requirement from Raw Water Storage Tanks I and II; but contains a minimum volume specification of 180,000 gallons based on an earlier FPL Fire Protection Technical Specifications commitment in the FPL letter L-77-390 to the NRC dated December 22, 1977. Requirements for the Fire Water Supply System have subsequently changed to provide a redundant minimum dedicated fire water supply of 300,000 gallons and the plant was modified to dedicate 300,000 gallons each of fire water from Raw Water Storage Tanks I and II; but the Current Technical Specifications were not updated to reflect the current plant condition.

<u>Safety Evaluation Summary</u>

Based on what is contained in the safety evaluation, the existing plant condition does not represent an unanalyzed condition, does not result in an unreviewed safety question or require changes to the Technical Specifications. This existing plant condition does not adversely affect plant safety and operation.

Issued: November 7, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-101 Revision 0

APPENDIX R HVAC MANUAL ACTIONS

This Safety Evaluation justifies and documents the acceptability of performing manual actions in response to fire damper closure in plant areas as a result of a postulated Appendix R fire. Fire dampers in HVAC ducts which close in response to a fire in that fire area may cause inadequate ventilation in other fire areas. To compensate for the closure of these fire dampers, manual actions are required to reestablish ventilation in areas not affected by the fire in order to ensure that safe shutdown can be achieved. The required manual actions have been previously transmitted to the site.

This Safety Evaluation addresses plant ventilation systems which are required to support Safety Related Equipment at Turkey Point Units 3 and 4 and is classified as Safety Related.

Safety Evaluation Summary

Based on the discussions contained in this evaluation, the performance of manual actions required to mitigate the consequences of fire damper closure as a result of a postulated Appendix R fire is acceptable with respect to plant safety. This conclusion is justifiable since the manual actions are required to ensure safe shutdown subsequent to a plant fire. These manual actions do not constitute an unreviewed safety question or require any changes to the Current, Interim, or Revised Technical Specifications. Therefore, prior NRC approval was not required.

Issued: October 3, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-109 Revision 0

INVERTER ROOMS HALON DILUTION

Halon 1301 systems are provided as automatic suppression in areas where water is not a preferred extinguishing agent. In order to adequately suppress a fire and to prevent re-ignition, the concentration of Halon within an area must be maintained at a certain percentage and for a minimum period of time. To maintain this concentration within the fire area in the event of Halon system actuation, forced air ventilating systems should be isolated to prevent dilution of the suppressing agent.

A Halon suppression system is provided at Turkey Point Units 3 and 4 in the Inverter Rooms (Fire Zones 108A and 108B). The Inverter Rooms are provided with forced ventilation which, in the event of Halon system actuation in these zones, is isolated by the automatic actuation of isolation dampers. However, the fans for the ventilation system remain running against the closed isolation dampers which could result in leakage across the dampers. Therefore, the concern exists that the Halon suppression system, upon actuation and discharge of Halon 1301, may be diluted to a concentration below the concentration required to extinguish and prevent re-ignition of a fire.

A calculation has been performed assuming continued operation of the HVAC systems for these Fire Zones. The calculation demonstrated that the Halon concentration within the Inverter Rooms is above the concentration and duration recommended by NFPA-12A. Therefore, although the present Halon system's design for the Inverter Rooms does not satisfy licensing criteria in the UFSAR, Appendix 9.6A, calculation results indicate that the concentration guidance provided in NFPA-12A for solid surface fires is met. Thus, a fire can be effectively extinguished within the Inverter Rooms even though the HVAC systems remain operating.

Since this safety evaluation addresses ventilation equipment required to support safety related components, this safety evaluation is classified safety related.

Safety Evaluation Summary

Based on the discussions contained in this evaluation, the Halon Suppression System associated with the Inverter Rooms is adequate to perform the intended function during and following a postulated Appendix R fire. In addition, Halon dilution in the Inverter Rooms (Fire Zones 108A and 108B) does not result in an unreviewed safety question or a reduction in the margin of safety as defined in the Technical Specifications, nor have an adverse affect on plant safety or operation.

Issued: November 3, 1989

This page was intentionally left blank.

SAFETY EVALUATION: JPN-PTN-SEMJ-89-111 Revision 1

ICW PIPING INSPECTION AND REPAIR

PCA Engineering, a corrosion engineering firm, performed an internal crawl-through inspection, including appropriate patching and/or repairs, of the Unit 3 Intake Cooling Water (ICW) system piping during the 1990 Unit 3 refueling outage.

The inspection and repair work was accomplished in accordance with Procedure 18712-296-MOO1 with Unit 3 in Mode 6 or with the reactor defueled (no mode) and all the spent fuel stored in the Spent Fuel Pool (SFP). To ensure Residual Heat Removal (RHR) and SFP cooling requirements were met, ICW pump operations were controlled in accordance with the applicable Technical Specifications and system operating procedures. Similar guidelines were used previously for inspection and patching/repair work conducted in the Unit 4 ICW system piping and, as such, do not constitute a change in operating practices. All activities associated with the inspection, cleaning, patching and/or repairs specified in Procedure 18712-296-MOO1 are classified as Safety Related; therefore, this Safety Evaluation is classified as Safety Related.

Safety Evaluation Summary

Because Procedure 18712-296-MOO1 assures that the RHR system and the SFP will be maintained in a safe condition, the inspection, cleaning, patching and/or repair work to be performed in the Unit 3 ICW system piping does not adversely impact plant safety and operation, does not constitute an unreviewed safety question or require a change to the Current, Interim or Revised Technical Specifications. Therefore, prior NRC approval was not required for all activities associated with the 1990 Unit 3 refueling outage ICW system piping crawl-through inspection.

The procedure for performing the ICW system inspection, cleaning, and repair work and the associated plant configuration were reviewed against the requirements of 10CFR50.59 and found to be acceptable; thus prior NRC approval was not required for implementation. This acceptability is based on the fact that operation in accordance with Procedure 18712-296-MOO1 and this Safety Evaluation does not constitute an unreviewed safety question or require a change to the Technical Specifications.

Issued: January 31, 1990

SAFETY EVALUATION JPN-PTN-SEMJ-89-115 Revision 1

DOCUMENTATION CHANGES TO PROCEDURE 0-ONOP-074.1

This safety evaluation analyzes the affect of a typographical error in Plant Procedure O-ONOP-074.1, "Standby Steam Generator Feedwater System Operation with Loss of Offsite Power and Loss of Auxiliary Feedwater". Auxiliary Feedwater Steam Supply Valves MOV-3-1403, 1404 and 1405 were incorrectly identified as physically being in Fire Zone 89 when in actuality they are in Fire Zone 84. Procedure O-ONOP-074.1 repeated this error. The discrepancy was investigated to ensure there was no adverse impact to the plant. The investigation confirmed that the design configuration and operation of the plant has not been affected by this typographical error.

Since the typographical error evaluated in this Safety Evaluation addresses a fire zone number discrepancy as associated with fire protection features, this evaluation has been classified Quality Related.

Safety Evaluation Summary

The proposed changes will correct documentation only. No design bases will be changed. No plant modification will be performed as a result of this change. FPL Procedure 0-ONOP-074.1 should be revised (steps 1.2, 2.3.2, 4.2, 4.3, 5.1.2.4 and 5.2) to indicate that the Unit 3 AFW steam supply valves are located in Fire Zone 84. Based on the above 10CFR50.59 safety evaluation, the correction of discrepancies in plant Procedure ONOP-074.1 will not result in an unreviewed safety question or require any changes to the Technical Specifications. In addition, these changes will not adversely affect plant operation and safety . Therefore, prior NRC approval was not required to implement these changes. Changes to the description in the FSAR are being addressed separately.

Issued: October 27, 1989

SAFETY EVALUATION JPN-PTN-SEMJ-89-121 Revision 0

SPECIAL TEST 89-01: FI-LPT POST INSTALLATION TEST

Special Test 89-01, Revision 0 involves two phases of testing.

The first phase of testing (Test Runs 1, 2, 4 and 5) involved testing of the Unit 4 secondary plant to determine the post-Fully Integrated Low Pressure Turbine modification heat balance. This comparison is necessary to validate the predicted performance increase of at least 8.1 MWe as guaranteed for this modification by the contract with Westinghouse.

The valve alignments for the first phase of testing (Special Test 89-01, Appendix D, attached) are required to duplicate the controlled steam and feedwater flow path provided in the pre-modification test, Test Procedure TOP 464. As in TOP-464, steam generator blowdown and Unit 4 auxiliary steam will be isolated. Auxiliary steam may be obtained from Unit 3, if available. The secondary plant was monitored closely during the testing to ensure system stability and proper plant operation. The testing was interrupted or terminated any time operations personnel considered it necessary to maintain plant stability.

The second phase of testing (Test Runs 3, 6, and 7) involved testing the secondary plant to determine the peak plant performance. The valve alignments for the second phase of testing were required to reduce steam leak paths as much as possible and without jeopardizing plant safety while minimizing effects on plant operation. Steam generator blowdown and Unit 4 auxiliary steam were isolated during testing. Auxiliary steam may be obtained from Unit 3, if available. The secondary plant was monitored closely during the testing to ensure system stability and proper plant operation. The testing was interrupted or terminated any time operations personnel consider it necessary to maintain plant stability.

The instrumentation identified in Appendix A of Special Test 89-01, is required to determine the performance of the turbine and the moisture separator reheaters. The test instrumentation was installed using tees or instrument manifolds on existing instrumentation lines. The test instrumentation was installed in parallel with instrumentation used for normal plant control and was also installed in parallel with safeguards instrumentation. The installation procedures in ST 89-01 provided alignment which avoided any spurious signals from occurring during testing alignment from affecting control of normal plant operations. The plant protection equipment was not adversely affected by this temporary system alignment. The test equipment did not adversely impact seismically installed or safety related equipment. Further, test instrumentation to be connected in parallel with safety

related instrumentation was qualified to maintain the pressure boundary during a seismic event and was seismically supported in accordance with Engineering approved (by calculation) temporary support detail drawings. Also, the connecting tubing was installed and supported in accordance with Specification 5177-J-711. This ensured the pressure integrity of the safety related equipment is not compromised.

Safety Evaluation Summary

Based on the above evaluation, the temporary system alteration required for the performance of this test, as described in Special Test 89-01, Revision 1 (attached), does not constitute an unreviewed safety question in accordance with 10CFR50.59.

Issued: November 7, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-90-048 Revision 1

MOV-843 A & B, MOV-744 A & B AND MOV-6386 STROKE TIMES

In response to changes in recorded valve stroke times, Engineering was requested to identify the time requirements for the system realignment functions performed by MOV-843 A&B, MOV-744 A~B and MOV-6386. This safety evaluation provides the current realignment times and demonstrates that neither an unreviewed safety question nor any technical specification change is present.

As part of PC/M 88-480, the limit switch setting for various motor operated valves have been changed to better reflect the actual position of the valve. This was accomplished by moving the switch settings closer to where the valve was actually opened or closed. Though no changes were made which would effect actual valve speed (i.e., no changes in motor speed or valve operator gearing), the valve stroke times, as measured by the change in the valve position indicator light, has increased. In response to changes in valve stroke times, Engineering has been requested to identify the time requirements for the system realignment functions performed by MOV-843 A&B, MOV-744 A&B and MOV-6386. These requirements define system functionality and should not be confused with equipment performance limits.

It should be noted that changes in measured stroke times due to limit switch adjustments does not necessarily mean a change in valve performance. As noted above, adjustments on valve limit switches do not impact valve opening or closure speed (i.e., no changes have been made to the valve motor operators or it's gearing) but they do impact the on/off status of the valve position indicator lights. While actual valve opening and closing speeds remain the same, stroke time measurements utilizing valve position indicator lights may change. The techniques utilized for measuring valve stroke times will not be evaluated in this safety evaluation.

Safety Evaluation Summary

Based on the evaluation, the response times for ECCS valve functions (at MOV-843 A&B and MOV-744 A&B) and Penetration 25 containment isolation valve function (at MOV-6386) does not represent an unreviewed safety question and changes to the plant Technical Specifications are not required.

Issued: April 19, 1990

SAFETY EVALUATION: JPN-PTN-SEMJ-90-054 Revision 0

CCW HEAT EXCHANGER STRESS CORROSION CRACKING AND ICW CHEMICAL INJECTION

A chemical injection program was initiated on the tube side of the Unit 3 CCW heat exchangers in July, 1989 as a test to determine the feasibility of using chemicals to mitigate heat exchanger fouling. The test demonstrated that chemical injection successfully maintained the CCW heat exchangers free from fouling, resulting in significant reductions in both Technical Specification LCO hours and maintenance work.

During the 1990 refueling outage, a 100% tube inspection was performed on each of the Unit 3 CCW heat exchangers using eddy current testing (ECT) methods. Results of the inspection revealed that 101 tubes exhibited cracking just inside the tubesheet at the ICW inlet end of each Unit 3 heat exchanger.

Following the discovery of the tube cracks, a root cause investigation was launched by the Equipment, Service and Inspection (ESI) staff. The results of the investigation reflected ESI's opinion that nitrogen in some form (as ammonia or nitrites) was responsible for the cracking. However the specific source of nitrogen could not be identified.

This SE provides justification for the continuation of chemical injection under 10CFR50.59 and investigates possible causal connections between chemical injection and stress corrosion cracking. The use of chemical injection in light of the stress corrosion cracking (SCC) does not adversely affect plant safety or operation. However, certain operational considerations will be required to manage the stress corrosion phenomena.

Safety Evaluation Summary

Although no absolute root causes has been identified for the SCC discovered on the Unit 3 CCW heat exchangers, it seems most likely that the increasing concentrations of ammonia in the canals coupled with the decomposition of microbes due to the biocide has allowed the CCW heat exchangers to cross the threshold of SCC susceptibility. Even with many of the tubes cracked 100% throughwall, there was no leakage or a breach in the tube pressure boundary and therefore no impact on the CCW heat exchangers ability to accommodate an accident. Since SCC is a delayed failure process, a suitable eddy current inspection program will identify any further tube cracking due to SCC prior to fracture. Even considering multiple tube fractures, such as might occur during a seismic event, the tolerance within the tubesheet would significantly limit any movement of the fractured tube

thereby substantially limiting the loss of CCW inventory. In this event, measures are in place to identify the source of leakage and implement appropriate actions to restore the integrity of the system.

Chemical injection coupled with an appropriate inspection program promotes the reliability of the system by minimizing or eliminating the need to remove the heat exchangers from service for frequent mechanical cleaning. With an inspection program and any subsequent corrective actions, SCC would be limited to the CCW heat exchanger tubes at the rolled sections since they exhibit the greatest residual stresses and are therefore the most susceptible to SCC. Therefore, assurance is provided that the occurrence of SCC does not occur at any other location other than at the rolled tube sections.

Since the intent of the inspection program is to identify any signs of SCC prior to tube fracture so that corrective actions such as tube replacement or plugging can be implemented, the operational capability of the CCW heat exchangers remains intact thereby ensuring that they will be capable of accommodating a design basis accident. Additionally, with chemical injection the operating margin of the CCW heat exchangers is maximized by continually eliminating the fouling tendency.

Based on the above, the heat exchanger tubes exhibiting stress corrosion cracking will not create an unreviewed safety question, and will not impact the Technical Specifications. Operation of the CCW heat exchangers which are susceptible to SCC does not constitute a safety concern since the enhanced inspection program will identify any cracked tubes prior to tube fracture, and these tubes will be subsequently plugged.

Additionally, the following activities have been initiated for the long term management of the CCW heat exchangers:

A. Eddy Current Test (ECT) 100% of all three heat exchangers, per unit, at least each refueling outage and prior to eddy current testing, the heat exchangers will be cleaned to ensure consistent test data.

B. Pursuit of an aggressive retubing program during refueling outages based on eddy current test data.

C. Replacement of all tubes with indications of circumferential cracks and exhibiting throughwall indications greater than 60% with new tubes.

D. Implementation of a monitoring program of the cooling canal water system for various parameters such as ammonia, pH, turbidity, etc that could contribute to SCC phenomena.

E. Review and optimization of the injection rates of sodium hypochlorite (NaOCl) and the corrosion inhibitor (Betz Industrial, Copper-trol CU-1) utilized with the chemical injection program.

Issue: June 1, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-89-061 Revision 0

TEMPORARY SYSTEM ALTERATION FOR THE CLEANING OF THE TURBINE LUBE OIL

A Temporary Westinghouse filtration system was installed to reduce particulates in the Unit 4 Turbine Lube Oil Reservoir. This temporary system was installed under TSA-4-90-87-10 and operated in accordance with Temporary Procedure TP-628.

The Technical Specifications do not directly address the Turbine Lube Oil System. This temporary system alteration for cleaning the turbine generator lube oil reservoir may be performed during any mode of reactor operation and will not affect the Technical Specification requirements.

Safety Evaluation Summary

The evaluation demonstrates that the temporary system alteration does not involve an unreviewed safety question or change to the Technical Specification pursuant to 10CFR50.59, and prior NRC approval for this activity was not required.

Issued: May 23, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-007 Revision 0

ADDITION OF CHECK VALVE IN LOOP B HOT LEG INJECTION HEADER

At Turkey Point Unit 4, each high head hot leg safety injection header contains a check valve that provides RCS pressure boundary isolation. These valves are required to be relatively leak tight to prevent reactor coolant out leakage. As a result, the valves must be periodically leak tested.

The Turkey Point Unit 4 Loop B hot leg Injection header check valve (4-874B) was leak tested and found not to leak in excess of what is allowed by Technical Specification 3.16. However, the leakage was greater than previously recorded. As a result, Florida Power and Light (FPL) has proposed installation of an additional check valve in the Unit 4 loop B hot leg injection header.

Safety Evaluation Summary

The check valve may be added to Unit 4's Loop B hot leg injection should the leak rate approach the TS limit while being tested in Mode 5. If added the check valve will not significantly impact the ECCS's performance. There are no plant restrictions as a result of the proposed modification to Turkey Point Unit 4. Therefore, the ECCS will be able to perform its safety-related function of providing sufficient emergency core cooling flow following a LOCA.

Issued: January 10, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-008 Revision 0

TEMPORARY SHIELDING REQUEST NO. 90-13

Temporary Lead Shielding was attached to the Unit 3 Steam Generator handholes inside containment at Elevation 30'-6" to protect maintenance personnel from excessive radiation exposure generated when the Steam Generator handhole covers are removed.

This safety evaluation concluded that the installation of the temporary lead shielding in accordance with conditions specified in the safety evaluation has no adverse impact on plant safety or operation and does not affect any Technical Specifications.

Safety Evaluation Summary

No existing safety analysis is affected and no new failure modes are introduced. As determined in the Safety Evaluation, this work does not constitute an unreviewed safety question or require a change to any Technical Specifications pursuant to 10CFR50.59.

Issued: February 6, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-020 Revision 0

TEMPORARY LEAD SHIELDING FOR REQUEST NO. 90-11

Temporary Lead Shielding was installed on a portion of the Unit 3 Safety Injection System, specifically, the piping from RCS Cold Leg A to 3 feet upstream of the check valve 875-A. Additionally, Health Physics requested shadow shielding adjacent to the pressurizer spray line on RCS Cold Leg Loop A.
The shielding was required to reduce personnel radiation exposure while performing maintenance on the check valve.

The temporary Lead Shielding was installed while Unit 3 was defueled and was removed prior to entering Mode 6.

This Safety Evaluation concluded that the installation of Temporary Lead Shielding has no adverse impact on plant safety or operation and does not affect any Technical Specification.

Safety Evaluation Summary

Since no existing safety analysis is affected and no new failure modes are introduced as demonstrated in the Safety Evaluation, this work does not constitute an Unreviewed Safety Question or a change to any Technical Specification pursuant to 10CFR50.59. Therefore, prior NRC approval was not required for implementation of the Temporary Lead Shielding request.

Issued: February 6, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-022 Revision 0

USE OF THE TRI-NUCLEAR UNDERWATER FILTRATION/VACUUM SYSTEM

Tri-Nuclear Corp's Portable Underwater Filter and Vacuum Units were used for filtration of the reactor cavity water during refueling operations. This increased water clarity and prevented delays due to poor visibility. These portable units were installed during Mode 6 and removed prior to entering Mode 4 and are not considered permanent plant equipment. The equipment is not safety-related (NSR), since it is not considered a part of the reactor coolant system pressure boundary, necessary to assure shutdown capability of the reactor, or to mitigate the consequence of an accident. The evaluation showed that the use of the underwater filter and vacuum units does not affect any Technical Specifications or Plant Operation or Safety.

Safety Evaluation Summary

The evaluation as contained within demonstrates that the subject activity does not involve an unreviewed safety question or change to the Technical Specifications pursuant to 10CFR50.59. Prior NRC approval for this activity was not required.

Issued: January 31, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-034 Revision 0

TEMPORARY LEAD SHIELDING REQUEST NO. 90-20

Health Physics has requested a review of existing temporary lead shielding for Units 3 & 4 cask wash areas (outside containment). The lead shielding was installed by H.P. in 1981 to reduce general area dose rates to the workers performing maintenance in the area around the spent fuel pit demineralizes. The high dose rate is generated by the spent fuel pit demineralizes.

The installation of the temporary lead shielding has been evaluated to ensure that there will be no detrimental effect on plant operation and safety and does not affect any Technical Specification based on the requirements stipulated herein. No mode restrictions are imposed with this safety evaluation.

Safety Evaluation Summary

This safety evaluation concludes that the temporary lead shielding does not require any changes to the Plant Technical Specifications and does not affect plant safety or operation pursuant to 10 CFR 50.59. Therefore, prior NRC approval was not required for implementation of this Shielding Request.

Issued: March 22, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-038 Revision 0

SPENT RESIN TRANSFER PIPING SPOOL SUPPORTS (TSA NO. 3-90-61-22)

An existing 1 1/2" diameter pipe spool piece at the cask station inside the Radwaste Building will be used to transfer spent resin. The spool piece currently is not properly supported.

This safety evaluation provided the requirements to temporarily support the pipe spool piece under TSA No. 3-90-61-22 and concluded that there is no adverse impact on plant safety or operation and does not affect any Technical Specification.

The spent resin transfer piping located inside the Radwaste Building does not perform a Safety Related function. Based on the walkdown of the piping configuration, there are no Safety Related systems, structures, or components located in the vicinity of the piping which could be adversely affected if the piping carrying radioactive spent resin collapsed under a postulated seismic event. Review of Regulatory Guide 1.143 indicates that liquid radwaste systems do not need to be designed to meet seismic criteria, which is consistent with UFSAR Section 11.1, and as a result, the pipe supports for the pipe spool are classified as Not Safety Related.

Safety Evaluation Summary

No existing safety analysis is affected and no new failure modes are introduced, as determined in the safety evaluation; therefore, this work did not constitute an unreviewed safety question or require a change to the Technical Specifications.

Issued: March 9, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-041 Revision 0

ACCEPTABILITY OF AS-FOUND CONDITION FOR RHR CHECK VALVE

The Unit 3 Residual Heat Removal System (RHR) "A" Pump Discharge Check Valve 3-753A was found to have a 1 1/32" long by 1/16" wide linear indication the full distance across the valve seat and extending into valve body (Attachment 1). A flaw evaluation was performed in accordance with general flaw evaluation methods contained in ASME Section XI.

This evaluation has determined, based on ASME Section XI flaw acceptance criteria, that this linear indication will not prevent the check valve from performing its intended function (backflow prevention through RHR pump while maintaining RCS pressure boundary integrity, Reference 1, page 9.3-9) for the time period between the issuance of this document and the start of the Diesel/Security Project Outage. Considered in this evaluation is the requirement for accident and post accident operation, and the possibility for several} shut downs.

This check valve is nuclear safety related (Reference S) as the RHR system is necessary to assure integrity of the RCS pressure boundary, to shutdown and maintain the reactor in a safe shutdown condition, and mitigate the consequences of an accident.

Safety Evaluation Summary

This evaluation shows that the continued use and operation of check valve 3-753A in its present condition between issuance of this document and the dual unit outage does not effect any Technical Specifications, Plant Operation or Safety, and that an unreviewed safety question does not exist.

Issued: March 21, 1990

SAFETY EVALUATION: JPN-PTN-SENJ-89-065 Revision 2

EPS ENHANCEMENT REPORT, HHSI PUMP SAFETY EVALUATION ON EPS ENHANCEMENT PROJECT TECH SPEC REVISION

Florida Power and Light has proposed an enhancement to the Emergency Power System (EPS) for the Turkey Point Nuclear Units 3 and 4, which includes the addition of two Emergency Diesel Generators (EDGs) with an enhanced electrical bus configuration. In order to implement this reconfiguration of emergency safety-related power supplies and buses, various proposed changes in plant Technical Specifications have been identified.

Among the proposed EPS Enhancement Project Technical Specifications is a change to the operability criteria for the HHSI pumps. The proposed change to the HHSI pump Technical Specification (Section 3/4.5.2 of the Revised Technical Specifications) would allow any one HHSI pump (i.e., one of two associated with the shutdown unit) to be taken out-of-service for an indefinite period of time with no ACTION statement imposed on the operating unit. This provision of the proposed EPS Technical Specifications would apply to a unit during power operation (Modes 1, 2, or 3), while the other unit is in a shutdown condition (Modes 4, 5 or 6). This proposed EPS Technical Specification is structured on the premise that one of the two HHSI pumps associated with the shutdown unit may be considered to be an installed spare, with an indefinite allowable out-of-service time while the other unit is operating at power.

Since the proposed EPS Technical Specifications could have the potential for reducing the number of available HHSI pumps from four to three for significant periods during one unit power operation, an evaluation of the number of HHSI pumps under the requirements of 10 CFR 50.92 has been performed. This safety evaluation will demonstrate that operation of one unit with the HHSI system in a three pump configuration, after modifications are completed under the EPS Enhancement Project (Reference 2), does not involve a significant decrease in the margin of safety (as required by 50.92), and an overall increase in the margin of safety will be realized.

The existing emergency power distribution system configuration consists of two Emergency Diesel Generators (EDG) shared between Turkey Point Units 3 and 4. Under this configuration one EDG supplies power to the "A" 4160 volts AC (VAC) bus of each unit and the other diesel powers the "B" 4160 VAC buses on each unit. One HHSI pump motor is powered from each of these four 4160 VAC buses.

The modifications performed under the EPS Enhancement Project provide an emergency AC distribution system design which will make each unit more independent of the other unit's electrical system. Under the enhanced EPS system configuration, one EDG will be

assigned to power each of the four 4160 VAC buses. There would be no change in the bus connections powering each of the four shared HHSI pumps, and each HHSI pump motor would continue to be powered directly from its associated 4160 VAC bus. In addition, various control and sequencer logic changes are being implemented to ensure that the enhanced EPS configuration relating to the HHSI pumps will still meet the train electrical independence criteria required by the Turkey Point design basis. Replacement of the sequencers under the EPS Enhancement Project would no longer allow (as in the existing design) a single active failure in the sequencing logic to result in the loss of one train of Engineered Safety Features (ESF) equipment, including two HHSI pumps and one RHR pump.

Under the EPS Enhancement Project, no changes will be made in the physical locations of HHSI pumps or associated valves, and no changes will be made in the configuration or physical location of HHSI system piping. In the current and future arrangement of pumps, two Unit 3 HHSI pumps 3A and 3B are placed side-by-side in the same Class I structure (Auxiliary/Radwaste Building) compartment. Adjacent to this compartment is a second Class I structure compartment containing the Unit 4 HHSI pumps 4A and 4B. Dividing these two compartments is a partial length and partial height (about 8 feet) reinforced concrete wall.

Safety Evaluation Summary

Among the proposed EPS Enhancement Project Technical Specifications is a change to the operability criteria for the HHSI pumps. The proposed change to the HHSI pump Technical Specification (Section 3/4.5.2) would allow any one HHSI pump (i.e., one of two associated with the shutdown unit) to be taken out-of-service for an indefinite period of time with no ACTION statement imposed on the operating unit. Since this proposed change could have the potential for reducing the number of operable HHSI pumps from four to three for significant periods during one unit power operation, an evaluation of the number of HHSI pumps under the no significant safety hazards criteria of 10 CFR 50.92 has been performed. This evaluation demonstrates that plant design bases affected by the HHSI pumps may still be met (with no reduction in the availability of HHSI pumps on demand) with only three operable pumps after modifications are completed under the EPS Enhancement Project as discussed in Reference 2.

Based on the evaluation performed, the proposed EPS changes will not reduce the availability of HHSI pumps on demand and, consequently, the available margin of safety, i.e., existing regulatory criteria continue to be satisfied. The proposed reduction in the required number of HHSI pumps does not preclude the availability of the fourth HHSI pump for events requiring safety injection. Since it is most likely that the fourth HHSI pump will continue to remain operable, even while one unit is

shutdown, the enhanced EPS redesign will actually have the affect of introducing additional margins to safety limits beyond those now inherent in the licensed basis
for the HHSI system.

Issued: July 2, 1990

SAFETY EVALUATION: JPN-PTN-SENJ-89-084 Revision 0

REACTOR VESSEL MISSILE SHIELD REMOVAL IN MODE 4 FOR RPI MAINTENANCE

The Unit 4 Missile Shield was removed in order to adequately investigate problems that existed with the rod position indicator. The CRDM Missile Shield was moved in accordance with plant procedures, with the additional restriction that heavy load movement over the steam generator, main steam and main feedwater piping was not permitted.

<u>Safety Evaluation Summary</u>

This activity did not involve an unreviewed safety question nor did it adversely impact the basis of any Technical Specification; therefore, prior NRC approval was not required. No FSAR revision was required.

Issued: August 14, 1989

SAFETY EVALUATION: JPN-PTN-SENJ-89-130 Revision 1

CONTROL ROOM SAFETY INJECTION BLOCK SWITCHES

Westinghouse informed FPL by a letter dated October 26, 1989 of a potential single failure design deficiency associated with the Control Room Safety Injection Block Switches (OT-2). Nuclear Engineering evaluated this concern and determined that it is applicable to Turkey Point. Bases have been provided to permit continued plant operation until plant changes can be implemented. This issue was reported under LER 250-89-018. This issue is considered to have relatively small safety significance due to its low probability of occurrence.

Revision 1 of this safety evaluation incorporates additional information which addresses the potential for latent switch failures which may affect the use of the EOPs. The results and conclusions of this safety evaluation are unaffected by this revision.

Safety Evaluation Summary

The safety evaluation identified a potential single failure with the control room Safety Injection block switch that was reportable under 10CFR50.73. Procedural changes and operator training have been identified which mitigate the potential for the existence of the postulated failure. These procedural changes have been determined not to involve an unreviewed safety question and require no changes to the technical specifications.

Issued: December 14, 1989

SAFETY EVALUATION: JPN-PTN-SENJ-90-045 Revision 2

SAFETY EVALUATION TO ADDRESS THE CONTAINMENT SUMP SCREEN DESIGN

As a result of a recent NRC IE Notice 88-77 "Debris In Containment Emergency Sump and Incorrect Screen Configurations," FPL performed walkdowns of the Turkey Point Units 3 and 4 containment sump screens against the design drawing illustrated in Figure 6.2-2 of the FSAR. FPL personnel performing the walkdown identified that the dimensions of the sump screens were as follows: (1) Unit 3 south sump - 5' by 5'; (2) Unit 3 north sump - 4'-4.5" by 3'-9.5"; (3) Unit 4 south sump - 5' by 5'; and (4) Unit 4 north sump - 4'-4" by 3'-9". Figure 6.2-2 from the Turkey Point FSAR shows that the dimensions of the sump screens are 8' by 8'. Based on this difference between the actual and FSAR dimensions, non-conformance reports were prepared to identify this inconsistency.

It has been determined that adequate NPSH is available to the Residual Heat Removal, SI and Containment Spray pumps with the current installed containment sump screens. Approach velocities in the near sump region are maintained low such that debris will not be transported to the screens. Further, the overall screen dimensions have been assessed and determined to be adequate.

The installation of smaller dimension containment sump screens will not prevent the ECCS from providing adequate core cooling capability during the post-accident recirculation mode of operation. Adequate NPSH is maintained under postulated recirculation conditions of one sump 100% blocked and the other sump 50% blocked.

Safety Evaluation Summary

Based on the safety evaluation, the as-built configuration of the containment sump screens does not represent an unreviewed safety question and changes to the plant technical specifications are not required.

Issued: May 14, 1990

SAFETY EVALUATION: JPN-PTN-SENS-89-086 REVISION 0

CONTINUOUS H2 MONITORING OF THE UNIT 4 RCS FOR EPRI RESEARCH

A temporary in-line reactor coolant probe was installed to continuously monitor reactor coolant for pH, redox potential and dissolved hydrogen for a period of approximately two weeks. The temporary in-line probe may be installed to continuously measure RCS parameters in the letdown line in any mode of reactor operation and will not adversely impact the operation or safety of the plant.

The RCS probe, and associated hardware are installed in the sample hood downstream of Sample Valve 971F which is classified as Not Safety Related. All other valves within the sample hood are classified as Not Safety Related.

This evaluation shows that the installation of the RCS probe does not affect the Technical Specification for identified leakage from the RCS or offsite dose rates from the release of airborne activity.

Safety Evaluation Summary

Since no existing safety analyses are affected and no new failure modes are introduced, this change did not constitute an unreviewed safety question or a change to any technical specification pursuant to 10CFR50.59.

Issued: September 13, 1989

SAFETY EVALUATION: JPN-PTN-SENS-89-096 Revision 0

PERFORMANCE OF THE BORAFLEX PANEL INTEGRITY ASSESSMENT PROGRAM

Turkey Point Unit 3 conducted a Boraflex Integrity Assessment to determine the integrity of the Boraflex panels used in Region I and Region II spent fuel storage racks.

The Boraflex Integrity Assessment Program was conducted in accordance with plant approved instructions. Plant procedures provide instructions for facilitating the set up and calibration of the testing equipment (in accordance with (IAW) vendor procedures) and the Blackness Testing of selected fuel rack cells (IAW vendor procedures).

Safety Evaluation Summary

This evaluation shows that the performance of the Blackness Testing does not affect any Technical Specification or plant operation. The Boraflex Integrity Assessment program is considered a test which was not previously evaluated in the SAR. Since no existing safety analyses are affected and no new failure modes are introduced, this test does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR 50.59.

Issued: October 10, 1989

SAFETY EVALUATION: JPN-PTN-SENS-89-112 Revision 1

TEMPORARY STORAGE OF A POWER RANGE NI DETECTOR INSIDE CONTAINMENT

A power range NI detector was replaced in Turkey Point Unit 4. In order to keep personnel dose as low as reasonably achievable (ALARA), it is desired to store the replaced detector inside containment.

This evaluation demonstrates that storage of a power range detector in the refueling cavity does not affect plant operation or any Technical Specifications. The evaluation requires removing the detector from the refueling cavity prior to flooding the refueling cavity.

Safety Evaluation Summary

Since no existing safety analyses are affected and no new failure modes are introduced, the temporary storage of the NI detector does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10CFR 50.59. Therefore, prior NRC approval was not required.

Revision 1 to this evaluation adds a discussion of the potential interaction of the NI detector with the refueling cavity sump drains. The conclusions of the evaluation are not affected.

Issued: December 21, 1989

SAFETY EVALUATION: JPN-PTN-SENS-89-135 Revision 0

PROCEDURE CHANGE TO OP-033, SPENT FUEL PIT COOLING SYSTEM

A change has been made for 3/4-OP-033, Spent Fuel Pit Cooling System, to allow draining the RWST through the Spent Fuel Pit demineralizer. This flow path to the waste holdup tank facilitates adjusting RWST boron concentration in order to comply with technical specification requirements.

This safety evaluation concludes that draining the RWST in this manner does not affect plant operation or any Technical Specifications. The evaluation requires the drain valve to be under operator control during the draining operation.

Safety Evaluation Summary

There are no technical specifications affected by the alignment of the RWST to the WHT through the SFP demineralizer drain valve. There is no change to technical specification 3.4.2 requirement for 320,000 gallons of water in the RWST in Modes 1-4. Thus, the evaluation demonstrates that the proposed activity does not involve an unreviewed safety question or change to the Technical Specification pursuant to 10CFR 50.59, and prior NRC approval for this activity was not required.

Issued: December 28, 1989

SAFETY EVALUATION: JPNS-PTN-89-4708

EVALUATION AND SAFETY EVALUATION OF TURBINE STOP VALVE OPERATION

On October 3, 1989, during performance of a test, the main steam turbine stop valve failed to travel within 1.0 inches of full closure when commanded. The main steam turbine lefthand stop valve closed within 1.0" as measured at the servovalve.

Nuclear Engineering has reviewed the anomaly with respect to the operation of the Turkey Point Unit 4 turbine stop valves. The turbine stop valves were found to not consistently close fully when they were cycled with the main steam line pressurized and with no steam flow. The valve was observed to remain open up to approximately 1 inch of actuator travel from fully closed. The stop valves have not been found to function abnormally in any other mode. When the valves do not fully close, the limit switches that provide input to the Reactor Protection System (RPS) and which provide control room indication of valve position do not close.

The Westinghouse Turbine Division has supervised tests of the turbine stop valves at the Turkey Point site and reviewed a test plan prepared by Turkey Point Plant management. Westinghouse has determined that although there may be some condition within the valve or actuator which prevents it from going full closed when there is no steam flow to assist closure, the effect is minor and if any steam flow were present it would assist in closing the valves and assure their closure. Based on successfully completing the test plan proposed by plant management, Westinghouse supports the determination that the valves can be considered operable for returning the Unit to service.

Safety Evaluation Summary

Westinghouse Nuclear Division has reviewed the function of the turbine stop valve position switch input to the RPS with respect to the design basis of the RPS and safety analyses. Westinghouse has concluded that these limit switches do not provide an essential input to the RPS and that the subject condition does not involve an unreviewed safety question or a change to the Technical Specifications.

Issued: October 6, 1989

SAFETY EVALUATION: SFB-7072

REPORT FOR REPLACING THE PLANT Q-LIST WITH THE TOTAL EQUIPMENT DATA BASE (TEDB)

Subsequent to the issuance of the Operating License for Turkey Point Units 3 and 4, FPL had established and implemented a Quality Assurance Program, as described in the FPL Topical Quality Assurance Report, which is in compliance with the requirements of Appendix B to 10CFR50 and approved by the NRC. The systems, components, and structures to which the FPL Topical Quality Assurance Report is applicable are set forth in the Turkey Point Units 3 and 4 Plant Q-List, which has been approved by the FPL Nuclear Engineering Department for use by plant personnel as a valid source for design information. The Turkey Point Plant Q-List is a computerized data base containing the current as-built information on a component level. FPL developed the Total Equipment Data Base (TEDB) in 1986 to expand the fields in the Plant Q-List, to add additional records for the same component, and to include non-engineering data for plant wide departmental use. The Plant Q-List and the TEDB have been continuously and concurrently updated to reflect the latest as-built plant configuration. Both data bases have been in parallel use for nearly two years. The TEDB Safety Classification Update Project further enhanced the TEDB by resolving differences between the Plant Q-List and the TEDB and providing the Safety Classification and basis for classification for over 20,000 items. Since the TEDB contains more information than the Plant Q-List, and as the TEDB has not dropped any required item in its data base from those required in the Plant Q-List, it is proposed to replace the Plant Q-List with the TEDB.

The proposed changeover to the TEDB will provide additional data to the users. The elimination of the Plant Q-List is cost effective as the IBM Mainframe computer which presently holds both data bases will be freed for other uses and the cost of maintaining the two data bases in parallel will be eliminated. The replacement of the Plant Q-List with the TEDB will also eliminate the inconsistencies and confusion involved in using the two data bases.

The Plant Q-List data base was electronically transferred to the TEDB and has been evaluated for satisfactory critical data match-up. Therefore, plant operation and safety are not affected. The changeover does not change any design or operating practice, or philosophy of the plant. There is no plant restriction associated with this changeover.

Safety Evaluation Summary

A stand-alone safety evaluation (Reference 8.9) was also performed to document the results of the evaluation on Licensing Requirements, Failure Modes and Effects Analysis, Plant Restrictions, Effect on Technical Specifications, and Unreviewed Safety Question Determination. A UFSAR Change Package was included with the safety evaluation. The Safety Evaluation concluded that the changeover from Plant Q-List to TEDB will not have an adverse effect on plant safety or operation, does not constitute an unreviewed safety question, and does not require any changes to the Current, Interim, and Revised Technical Specifications.

Therefore, there is a high confidence level in concluding that the TEDB can be used by the on-site and off-site plant personnel as a valid source for design information.

Issued: October 19, 1989

SECTION 3

Unit 3 Cycle 12 Core Load

SAFETY EVALUATION: JPN-PTN-SECS-89-110 Revision 0

TURKEY POINT UNIT 3 CYCLE 12 RELOAD SAFETY ANALYSIS REVIEW 10CFR50.59 DETERMINATION

The Turkey Point Unit 3 Cycle 12 core loading consists of 52 fresh Region 14 Optimized Fuel Assemblies (OFA), 96 previously burned OFA, and 9 previously burned Low Parasitic (LOPAR) fuel assemblies. Four of these LOPAR assemblies are reconstituted from Cycle 8 and contain a total of ten stainless steel filler rods and one low enriched fuel pin. This cycle design also incorporates 4016 Integral Fuel Burnable Absorbers (IFBA) and 96 Wet Annular Burnable Absorbers (WABA). In addition, 12 assemblies contain the reduced length hafnium absorber rods to reduce the neutron flux at the reactor pressure vessel belt-line weld. Twenty-four of the Region 14 assemblies contain 116 IFBA pins each, while twelve assemblies contain 60 IFBA pins, and sixteen assemblies contain 32 IFBA pins each. The IFBA coating covers 120 inches of active fuel centered on the assembly mid-plane. The WABA are arranged in 16 clusters containing 4 pins each and four clusters containing 8 pins. These WABA pins are 108 inches in length and are also centered within the active fuel region of the assembly. Finally, in two of the reconstituted LOPAR assemblies there are two clusters containing four pyrex rods each for a total of eight. These rods are full length (144 inches) centered within the assembly active fuel length. This core loading will provide a cycle length of 14,200 MWD/MTU including a 500 MWD/MTU power coastdown.

The Region 14 fuel assemblies are similar to the Region 13 fuel assemblies used in Cycle 11 except for the following: Debris Resistant Fuel Assembly (DRFA) design changes have been incorporated in the Region 14 fuel design to make the assembly more resistant to debris fretting failures. In addition, the Region 14 fuel incorporates axial blankets, extended burnup modifications, reconstitutable top nozzles, standardized fuel pellets, reduced fuel rod backfill pressures, 4g fuel rod plenum springs, and 304L stainless steel grid sleeve material. These modifications provide either improved fuel performance, fuel utilization or manufacturing reliability. These features have been reviewed and approved by the NRC and complies with all current nuclear regulatory requirements and have been shown to be compatible with the existing fuel assemblies in Turkey Point Unit 3.

Safety Evaluation Summary

Based on the technical evaluation/analyses performed by Westinghouse, it can be concluded that the Turkey Point Unit 3 Cycle 12 reload design meets all design criteria, is bounded by the results of the referenced analyses, and can be implemented with no changes required to the existing Turkey Point Unit 3 Technical Specification. Therefore, it can be stated that:

i. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

The Turkey Point Unit 3 Cycle 12 reload design does not change the overall configuration of the plant. The mode of operation of the plant remains unchanged. Changes to the fuel assembly design features, i.e., axial blankets, extended burnup modifications, reconstitutable top nozzles, standardized fuel pellets, and reduced fuel rod backfill pressures have been reviewed and approved by the NRC and have been shown to be compatible with existing fuel assemblies in Turkey Point Unit 3. Changes to the fuel assembly design to make the assembly resistant to debris induced fretting do not change the mechanical or thermal-hydraulic performance of the fuel assembly. The Small and Large Break LOCA events were re-evaluated against the Cycle 12 fuel assembly design changes and shown to meet the acceptance criteria. The Reload Safety Evaluation report demonstrates that the consequences of an accident or malfunction of equipment important to safety have not been increased beyond those evaluated in the previous analyses since all the transients meet current criteria. Therefore, the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, is not increased.

ii. A possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created.

The Turkey Point Unit 3 Cycle 12 reload design does not change the overall configuration of the plant, or the mode of operation of the plant. The changes to the design features have been evaluated/analyzed and have been shown to be compatible with the existing fuel assemblies in Turkey Point Unit 3. Therefore, a

possibility for a new accident or equipment malfunction has not been created.

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

The Turkey Point Unit 3 Cycle 12 reload design neutronics input and the resulting safety analysis has been reviewed, and in all cases the results are well within the acceptance criteria of the design basis. Based on FPL's independent review of the Reload Safety Evaluation report it can be concluded that the Turkey Point Unit 3 Cycle 12 reload design does not result in a reduction to the margin of safety relative to the Technical Specification basis for Turkey Point Unit 3.

SECTION 4

Annual Report of Power Operated Relief Valve (PORV) actuations

ANNUAL REPORT OF SAFETY AND RELIEF VALVE CHALLENGES

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

The following is a list of power operated relief valve (PORV) actuations for Turkey Point Units and 4 from July 1, 1989 to June 30, 1990.

Procedure Title Key

3-OP-041.4 and 4-OP-041.4	Overpressure Mitigation System
3-OSP-041.4 and 4-OSP-041.4	Overpressure Mitigation System Nitrogen Backup Leak and Functional Test
OP 0209.1	Valve Exercising Procedure

Unit 3

February 6, 1990	PORV 455C was cycled per 3-OSP-041.4
February 6, 1990	PORV 456 was cycled per OP 0209.1
April 7, 1990	PORV 455C and 456 were cycled per 3-OSP-041.4
April 7, 1990	PORV 455C and 456 were cycled per 3-OP-041.4
April 18, 1990	PORV 455C and 456 were cycled per 3-OP-041.4
April 25, 1990	PORV 455C and 456 were cycled per 3-OP-041.4
May 3, 1990	PORV 456 was cycled per 3-OSP-041.4
May 4, 1990	PORV 455C was cycled per 3-OSP-041.4

Unit 4

September 23, 1989	PORV 455C and 456 were cycled per 4-OP-041.4
September 23, 1989	PORV 456 was cycled per OP 209.1
September 27, 1989	PORV 455C and 456 were cycled per 4-OSP-041.4
September 27, 1989	PORV 455C and 456 were cycled per 4-OSP-041.4
September 27, 1989	PORV 456 was cycled per OP 0209.1
September 28, 1989	PORV 455C was cycled per 4-OP-041.4
September 28, 1989	PORV 455C was cycled per 4-OSP-041.4
September 28, 1989	PORV 455C was cycled per 4-OSP-041.4
September 28, 1989	PORV 455C was cycled per OP 0209.1
September 29, 1989	PORV 455C was cycled per 4-OP-041.4

SECTION 5

Unit 3 Steam Generator Tube Inspections

EDDY CURRENT EXAMINATION RESULTS					
PLANT: TURKEY POINT PLANT UNIT NO. 3					
EXAMINATION DATES: FEBRUARY 28, 1990 THRU MARCH 13, 1990					
STEAM GENERATOR NUMBER	TOTAL TUBES INSPECTED	TOTAL IND > OR = TO 20% TO 39%	TOTAL IND > OR = TO 40% TO 100%	TOTAL TUBES PLUGGED AS PREVENTIVE MAINT	TOTAL TUBES PLUGGED
3E210A	3203	17	1	3	4
3E210B	3205	22	2	3	5 (1)
3E210C	3194	32	2	4	6 (2)

(1) 4 Hot Leg plugs were also replaced
(2) 3 Hot Leg plugs were also replaced

LOCATION OF INDICATIONS

STEAM GENERATOR	AVB BARS	DRILLED SUPPORT 1 THROUGH 6		TOP OF TUBE SHEET TO 1 DRILLED SUPPORT	
		HOT LEG	COLD LEG	HOT LEG	COLD LEG
3E210A	5	5	4	2	2
3E210B	8	6	3	7	0
3E210C	20	6	4	2	2

STEAM GENERATOR TUBES PLUGGED								
STEAM GENERATOR 3E210A			STEAM GENERATOR 3E210B			STEAM GENERATOR 3E210C		
ROW	COLUMN	REMARKS	ROW	COLUMN	REMARKS	ROW	COLUMN	REMARKS
7	5	STUB TUBE	7	5	STUB TUBE	14	6	HOT LEG TUBE SHEET
7	13	STUB TUBE	7	13	STUB TUBE	40	38	#2 AVB
9	32	#4 SUP. HOT LEG	40	39	#5 SUP HOT LEG	35	47	#3 AVB
33	44	#3 AVB	41	32	#3 AVB	38	53	#3 AVB
			42	43	#2 AVB	38	54	#2 AVB
						13	89	COLD LEG TUBE SHEET
		THE FOLLOWING HOT LEG PLUGS WERE ALSO REPLACED						
			42	30		7	5	STUB TUBE
			25	32		7	13	STUB TUBE
			45	43		14	89	
			45	44				

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210A EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
9	2	37	HOT LEG	06H 2.9
16	4	29	HOT LEG	TSH 1.4
13	5	29	HOT LEG	TSH 3.8
7	10	37	COLD LEG	01C 10.2
22	15	30	COLD LEG	03C 45.0
10	16	34	COLD LEG	02C 23.9
6	18	29	COLD LEG	BAC 24.5
31	19	24	COLD LEG	04C 48.5
*33	44	39	AVB	AV3 .0
38	45	24	AVB	AV2 .0
22	52	38	HOT LEG	04H 40.9

HOT LEG (INLET)
COLD LEG (OUTLET)

* Preventively Plugged

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210A EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
15	55	30	COLD LEG	TSC 8.1
30	58	27	AVB	AV1 .0
		29	AVB	AV3 .0
28	59	20	AVB	AV2 .0
27	70	21	HOT LEG	01H 46.0
32	75	25	AVB	AV3 2.1

HOT LEG (INLET)
COLD LEG (OUTLET)

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210B EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
10	7	32	COLD LEG	02C 31.8
23	10	32	HOT LEG	06H 3.7
19	12	37	HOT LEG	TSH .5
28	14	29	HOT LEG	01H 9.8
37	23	25	COLD LEG	06C .0
28	28	38	COLD LEG	03C 34.2
6	32	32	HOT LEG	TSH 39.0
5	34	31	HOT LEG	TSH 31.3
42	37	37	HOT LEG	TSH .3
34	38	26	AVB	AV2 .0
		25	AVB	AV3 .0

HOT LEG (INLET)
COLD LEG (OUTLET)

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210B EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
42	38	28	HOT LEG	TSH 1.3
		27	HOT LEG	TSH 3.1
39	39	37	HOT LEG	05H .3
41	42	29	AVB	AV2 .0
*42	43	38	AVB	AV2 .0
		20	AVB	AV3 .0
6	44	38	HOT LEG	TSH 38.0
42	45	23	AVB	AV2 .0
34	46	24	AVB	AV3 .0
11	72	27	HOT LEG	02H 36.5
11	85	26	HOT LEG	02H 36.6

HOT LEG (INLET)
COLD LEG (OUTLET)

* Preventively Plugged

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210C EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
27	30	24	AVB	AV2 13.0
4	34	30	COLD LEG	TSC 28.4
*40	38	35	AVB	AV2 .0
		30	AVB	AV3 .0
		27	AVB	AV4 .0
8	39	33	COLD LEG	03C 11.2
33	39	33	AVB	AV1 .0
		25	AVB	AV3 .0
35	41	33	AVB	AV1 .0
		22	AVB	AV2 .0
33	43	21	AVB	AV3 .0

HOT LEG (INLET)
COLD LEG (OUTLET)

EDDY CURRENT EXAMINATION RESULTS

PLANT: Turkey Point Nuclear Power Plant Unit No. 3
STEAM GENERATOR: 3E210C

EXAMINATION DATES: February 28, 1990 THRU March 13, 1990

ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
35	43	21	AVB	AV1 .0
		22	AVB	AV2 .0
		27	AVB	AV3 .0
		22	AVB	AV4 .0
13	44	27	HOT LEG	02H 51.2
35	44	27	AVB	AV2 .0
		28	AVB	AV3 .0
35	45	34	AVB	AV2 .0
		22	AVB	AV4 .0
30	46	27	AVB	AV1 .0
		25	AVB	AV2 .0

HOT LEG (INLET)
COLD LEG (OUTLET)

EDDY CURRENT EXAMINATION RESULTS				
PLANT: Turkey Point Nuclear Power Plant Unit No. 3 STEAM GENERATOR: 3E210C EXAMINATION DATES: February 28, 1990 THRU March 13, 1990				
ROW	COLUMN	% TUBE WALL PENETRATION	ORIGIN	LOCATION
		21	AVB	AV3 .0
*35	47	22	AVB	AV2 .0
		39	AVB	AV3 .0
30	48	34	AVB	AV3 .0
14	53	22	HOT LEG	TSH 1.2
*38	53	34	AVB	AV2
		39	AVB	AV3 .0
*38	54	22	AVB	AV1 .0
		36	AVB	AV2 .0
		20	AVB	AV3 .0

HOT LEG (INLET)
COLD LEG (OUTLET)

* Preventively Plugged