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SUBJECT: Forwards util rept on "Changes,Tests & Experiments Made W/o Prior Commission Approval" for period 900701-910630.

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AUG 30 1991

L-91-214
10 CFR 50.59(b)(2)

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

Gentlemen:

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

Attached is Florida Power & Light Company's Report on "Changes, Tests and Experiments Made Without Prior Commission Approval" for the period July 1, 1990 through June 30, 1991. Should you have any questions please contact us.

Very truly yours,

T F Plunkett

T. F. Plunkett
Vice President
Turkey Point Nuclear

TFP/DPS/ds

Attachment

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

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TURKEY POINT PLANT UNITS 3 AND 4

DOCKET NUMBERS 50-250 AND 50-251

CHANGES, TESTS AND EXPERIMENTS

MADE AS ALLOWED BY 10 CFR 50.59

FOR THE PERIOD OF

JULY 1, 1990 THROUGH JUNE 30, 1991

INTRODUCTION

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

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

which are conducted without prior Commission approval be reported to the Commission at least annually. This report is intended to meet this requirement for the period of July 1, 1990, through June 30, 1991.

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

In Section 1, Plant Change/Modifications, the PC/M Classifications are coded as follows:

NNSR - Non Nuclear Safety Related

SR - Safety Related

QR - Quality Related

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SECTION 1

PLANT CHANGE/MODIFICATIONS

PLANT CHANGE/MODIFICATION 79-011-13

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

HEAT TRACING - CIRCUIT CHANGE OUT

Summary:

This Engineering Package provides for the replacement of the existing chromalox MI cable with chemelex type self-limiting cable. The chromalox heat tracing cable had a poor performance record and several reportable occurrences due to flow blockage from boron precipitation. The chemelex type cable has been used at the plant on a limited basis and has proven to be reliable and easier to install and maintain.

Safety Evaluation:

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

PLANT CHANGE/MODIFICATION 83-006-14

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	10/03/90

STATION BATTERY REPLACEMENT

Summary:

This Engineering Package provided for the replacement of the existing station batteries with equivalent lead calcium GNB batteries. The change was made at this time because the previous batteries were showing signs of aging (reduced margin).

Safety Evaluation:

This modification was a like for like change.

This modification does not have any adverse effect on the plant safety or operation. 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 84-070-15

PC/M CLASSIFICATION : QR
UNIT : 3/4
TURN OVER DATE : 12/21/90

POST ACCIDENT SAMPLING SYSTEM (PASS) LONG TERM MODIFICATIONS

Summary:

The changes described in this PC/M were performed in response to the Post Accident Sampling System (PASS) Long Term Modification Program. The purpose of these changes is to enhance system operations and provide higher on-line sample reliability for the PASS. Additional shielding has been added to lower the post-accident dose rates thereby making the areas more accessible.

This Engineering Package also provides for the documentation of the "as-installed" conditions of the PASS System. This change provides clarification of the position status of several PASS manual valves. The addition of a H_2 analyzer flow meter and pressure gage are also documented.

Safety Evaluation:

This modification clarifies the drawing without changing the system operation.

This modification does not have any adverse effect on the plant safety or operation. 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 84-215-02

PC/M CLASSIFICATION : NNSR
UNIT : 3
TURN OVER DATE : 12/17/90

INSTALLATION OF UNIT 3 C BUS UNDERGROUND TRANSMISSION LINE

Summary:

This Engineering Package provides for the installation of the 230 kV pipe cable for the Unit 3 C-bus transformer from the Unit 3 start-up transformer. No 230 kV connections were performed under this PC/M, only the laying of the conduit and cable.

This PC/M in conjunction with PC/M 84-137 will improve the availability of offsite power by providing a new power source to the Unit 3 C-bus Transformer.

Safety Evaluation:

The C bus is not safety related.

This modification does not have any adverse effect on the plant safety or operation. 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 84-216-03

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	12/17/90

INSTALLATION OF UNIT 4 C BUS UNDERGROUND TRANSMISSION LINE

Summary:

This Engineering Package provides for the installation of the 230 kV pipe cable for the Unit 4 C-bus transformer from the Unit 4 start-up transformer. No 230 kV connections were performed under this PC/M.

This PC/M in conjunction with PC/M 84-137 will improve the availability of offsite power by providing a new power source to the Unit 4 C-bus Transformer.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 85-032-02

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	02/01/91

4160 KV SWITCHGEAR CONTROL MODIFICATION

Summary:

This Engineering Package adds a breaker position interlock in the closing control circuit of each incoming power supply circuit breaker. This change allows separate testing of each breaker without interrupting the normal or alternate supply to the 4160 V 4C-bus. This change also modifies the breaker trip circuit to connect the red indicating light directly to the breaker trip coil and adds a diode to isolate the breaker failure scheme. In addition, the modification includes all the internal wiring necessary to accomplish the above changes.

Safety Evaluation:

This modification involves breaker internal wiring changes and the addition of diodes which have no effect on the equipment seismic response since diodes are static devices. The safety related function of the breaker is not affected. This modification enhances the reliability of the incoming transformer breakers.

This modification does not have any adverse effect on the plant safety or operation. 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 85-037-03

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	07/25/90

UPGRADE OF RECIRCULATION FLOW FOR THE UNIT 4 CONTAINMENT SPRAY PUMP

Summary:

This Engineering Package installs valve 4-896T in a short piping spool from the existing Unit 4 recirculation line to the suction of the "A" Containment Spray Pump (CSP). This will increase the flow to 400 gpm. The new valve has a special trim, compatible with the high flow rate and low downstream pressure. The valve trim's unique design enables pressure drop and recovery through the valve that precludes undesirable noise and cavitation.

Safety Evaluation:

From a design and safety standpoint, the new Unit 4 recirculation system is identical to the previously installed Unit 3 recirculation system. The new installed system meets or exceeds the design requirements as originally installed.

This modification does not have any adverse effect on the plant safety or operation. 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 85-102-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3
TURN OVER DATE	:	02/01/91

SUBSTATION 3AC INTERIM ENVIRONMENTAL DESIGN ENHANCEMENT

Summary:

This Engineering Package provides for bus insulation/sealant changes per the vendor's recommendations.

This PC/M covers design changes to non-safety related 5 kV switchgear/substation 3AC.

Safety Evaluation:

This change involves environmental design enhancements only.

This modification does not have any adverse effect on the plant safety or operation. 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 85-132-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3
TURN OVER DATE	:	09/19/90

MOISTURE SEPARATOR REHEATER MODERNIZATION

Summary:

This Engineering Package provides for the replacement of the existing moisture separator reheater (MSR) tube bundles with functionally equivalent, but more efficient tube bundles. This PC/M also provides for the installation of two Reheater Drain Tank drain line flow measuring instruments and test connection points and thermowell.

Safety Evaluation:

This modification is classified as non-nuclear safety related.

This modification does not have any adverse effect on the plant safety or operation. 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 85-134-04

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	07/03/90

CHARGING PUMP PACKING REPLACEMENT

Summary:

This Engineering Package provides for the replacement of the packing in the charging pumps. The new packing configuration is similar in design to packing used in other industries with similar performance requirements. The new packing will not affect the limits set by Technical Specifications. The new packing is equal to or better than the existing packing.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 85-156-02

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	08/27/90

MAIN STEAM CHECK VALVE CLAPPER NUT LOCKING PROVISIONS

Summary:

This Engineering Package eliminates the pins inserted axially between the main steam check valves (MSCVs) nut and stud attaching the disc to the tail link. The nut was welded to the stud to provide a more positive locking method. The nut material was changed to be compatible with the stud for welding.

The clearance required to allow the disc to seat properly, allows vibration. This vibration has worn out the standard nut locking devices in the past, with the exception of welding the nut to the stud. Welding was concluded to be the only positive means to assure nut retention.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 85-175-03

PC/M CLASSIFICATION : SR
UNIT : 3
TURN OVER DATE : 01/23/91

AUXILIARY FEEDWATER BACKUP NITROGEN STATION MODIFICATIONS

Summary:

This Engineering Package provides for an enhancement of the existing Auxiliary Feedwater (AFW) Backup Nitrogen System. Additional bottles were added to provide an additional reserve operating time.

The design basis of each Auxiliary Feedwater (AFW) Nitrogen Station is to support a minimum of two hours of AFW system operation without operator action. Three nitrogen bottles are valved on-line to meet this requirement. (Two hours has been judged to be sufficient time to restore steam generator level following a postulated loss of offsite power event.) When the pressure of the three bottles decreases to 650 psig, the low pressure alarm will alert the operators that the two spare (off-line) bottles must be valved in within 80 minutes after initiation of the alarm, in order to maintain automatic operation of the AFW Flow Control Valves. The two spare bottles are required to provide a minimum of two hours of operation in order to allow replacement of the three depleted nitrogen bottles.

Safety Evaluation:

This modification enhances the availability and reliability of the nitrogen supply to the AFW control valves and does not alter the basic function of the AFW system.

This modification does not have any adverse effect on the plant safety or operation. 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 86-024-02

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	07/11/90

ICW SYSTEM HEAT EXCHANGER ISOLATION VALVES

Summary:

This Engineering Package provides for the replacement of the ICW heat exchanger isolation valves (50-4-314 and 50-4-334), with spring-to-close, fail-closed, air operated valves that also have redundant safety injection system actuation. The replacement valves, POV-4-4882 and POV-4-4883, will be operated as manual valves. The instrumentation and controls to these valves will not be connected under this PC/M.

Safety Evaluation:

This PC/M maintains the previous operational capability of the ICW system for both normal and accident conditions. There are no changes in valve type, setting, or operation.

This modification does not have any adverse effect on the plant safety or operation. 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 86-077-02

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	01/23/91

STEAM GENERATOR WET LAYUP CONTAINMENT ISOLATION VALVE REPLACEMENT

Summary:

This Engineering Package provides for installation of spectacle flanges in the steam generator wet layup (SGWL) system downstream of the SGWL containment isolation valve. The new spectacle flanges will function as containment isolation and ISI code boundaries until the current containment isolation valves can be replaced.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-084-03

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	08/27/90

DEMINERALIZATION REGENERATION WASTE NEUTRALIZATION TANK INSTALLATION

Summary:

This Engineering Package provides for installation of a neutralization tank and supporting foundation. This tank is a component of the demineralizer regeneration waste neutralization system. The balance of the system will be addressed in PC/M 85-195. This tank will be used to store and neutralize water treatment plant demineralizer regeneration waste prior to discharging to the neutralization basin and/or Intake Canal.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-136-01

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	11/01/90

MAIN STEAM LINE VALVES - LIFTING DEVICES

Summary:

This Engineering Package provides for the removal of the lifting devices from the Main Steam Safety Valves (MSSVs). Removal of the lifting devices eliminates a potential failure mode for these valves. This modification is in response to NRC Information Notice 84-33.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-147-03

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	12/20/90

LOW PRESSURE TURBINE ROTOR REPLACEMENT

Summary:

This Engineering Package provides for the replacement of the Unit 3 Low Pressure Turbine Rotor. The change out of the turbine rotor is a like for like replacement of equipment.

The new turbine rotor replaced the older disc and shaft design with the latest available technology. The new rotor is a single machined extrusion which eliminates discs and keyways as well as other locations which are highly susceptible to stress corrosion cracking. In addition, crack growth is inhibited by the selection of materials which have lower ultimate strength and increased ductility.

To ensure that the Low Pressure Turbine Rotor will not catastrophically fail from stress corrosion cracking, the rotors should be inspected at the interval recommended by the Vendor. The vendor recommended interval is 40 years. As a good maintenance practice, it is FPL's intention to inspect the rotors at a 5 year interval.

Safety Evaluation:

There are no accidents or malfunctions of a different type from the safety analysis created by this modification. This modification does not have any adverse effect on the plant safety or operation. 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 86-148-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3
TURN OVER DATE	:	10/04/90

DISCONNECTION OF TIE BREAKER OVERCURRENT INTERLOCKS FROM 4160 V BUS LOCKOUT CIRCUIT

Summary:

This modification is a result of a Failure and Effects Analysis (FEMA) performed for the 4160 V switchgear bus lockout schemes, transmitted to Florida Power and Light August 12, 1986. The FEMA determined that in a loss of offsite power scenario, a failure of relay 174X/TDDO associated with any tie breaker would cause the loss and lockout of both A and B train 4160 V safety buses. To meet the single failure criteria, the overcurrent interlocks described above were disabled by a temporary system alteration.

This Engineering Package provides for the permanent disconnection of the tie breaker overcurrent interlocks from the 4160 V bus lockout circuit. This modification was required to eliminate the single failure concern during loss of off-site power events.

Safety Evaluation:

The overcurrent lockout schemes perform an equipment protective function and affect the 4160 V safety related buses only. Plant procedures will provide the protective function of the lockout circuits being deleted. Thus this modification will have no impact on the operability of the safety related buses.

This modification does not have any adverse effect on the plant safety or operation. 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 86-149-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	10/04/90

DISCONNECTION OF TIE BREAKER OVERCURRENT INTERLOCKS FROM 4160 V BUS LOCKOUT CIRCUIT

Summary:

This modification is a result of a Failure and Effects Analysis (FEMA) performed for the 4160 V switchgear bus lockout schemes, transmitted to Florida Power and Light August 12, 1986. The FEMA determined that in a loss of offsite power scenario, a failure of relay 174X/TDDO associated with any tie breaker would cause the loss and lockout of both A and B train 4160 V safety buses. To meet the single failure criteria, the overcurrent interlocks described above were disabled by a temporary system alteration.

This Engineering Package provides for the permanent disconnection of the tie breaker overcurrent interlocks from the 4160 V bus lockout circuit. This modification was required to eliminate the single failure concern during loss of off-site power events.

Safety Evaluation:

The overcurrent lockout schemes perform an equipment protective function and affect the 4160 V safety related buses only. Plant procedures will provide the protective function of the lockout circuits being deleted. Thus this modification will have no impact on the operability of the safety related buses.

This modification does not have any adverse effect on the plant safety or operation. 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 86-185

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	08/27/90

ADDITION OF ANNUNCIATION IN MAIN CONTROL ROOM ON LOSS OF EDG CONTROL POWER

Summary:

This Engineering Package provides annunciation in the main control room upon loss of Emergency Diesel Generator (EDG) control power. This modification will help detect the unavailability of any EDG due to loss of control power. This package also replaced the non-resistored full voltage indicating lights at EDG 3A engine panel 3C13 and EDG 3B engine panel 4C13 with resistored indicating lights in response to Licensee Event Report (LER) 85-002-00.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-195-02

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	10/31/90

ADDITION OF CONTINUOUS TUBE CLEANING CAPABILITY TO THE CCW HEAT EXCHANGERS

Summary:

This Engineering Package provides for the addition of a continuous tube cleaning capability to the CCW heat exchangers. The new cleaning system operates by introducing sponge rubber balls into the intake cooling water supply line for each component cooling water heat exchanger. The normal flow forces the balls through the tubes to maintain cleanliness. The balls are flexible enough to move around minor obstructions. Screens in the discharge lines collect the balls for reuse.

The CCW heat exchanger heat transfer capacity is being ensured by an on-going monitoring program in accordance with the corrective action indicated in the Justification for Continued Operation for ICW System Design, JPE-LR-87-045, Revision 1. Plant operating restraints are based on a monitoring program to ascertain heat transfer characteristics and not performance of a cleaning function.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-201-04

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 08/06/90

HIGH INITIAL RESPONSE (HIR) BRUSHLESS EXCITATION SYSTEM

Summary:

This Engineering Package provides for the upgrade of the high initial response brushless excitation system which allows the main generator to respond quickly to changes in system voltage.

This upgrade consists of the following modifications:

A. The existing permanent magnet generator (PMG) was replaced by a higher voltage PMG with a higher ampere rating;

B. The AC exciter stationary field winding was redesigned to obtain a lower time constant and lower voltage drop;

C. A laminated exciter frame and core was provided to reduce the time required for the magnetic flux to change in the AC exciter;

D. Changes were also required in the automatic voltage regulator such as larger power amplifier drawers.

Safety Evaluation:

The Turbine Generator does not perform any safety related function. The modifications to the Turbine Generator are classified as non safety related.

This modification does not have any adverse effect on the plant safety or operation. 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 86-203-02

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 08/02/90

HYDROGEN DETECTION SYSTEM ADDITION TO TURBINE GENERATOR EXCITER AND COUPLING HOUSINGS

Summary:

This Engineering Package provides for the addition of a hydrogen detection system to the turbine generator exciter and coupling housings.

The hydrogen detection system is designed to detect a hydrogen gas buildup in the exciter housing before the gas concentration reaches the Lower Explosive Limit (LEL) and to alert the operators of this situation.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 86-238-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	06/05/91

SAFETY RELIEF VALVE REPLACEMENT, COMPONENT COOLING WATER SYSTEM

Summary:

This Engineering Package provides for the replacement of safety relief valves on the Component Cooling Water system. This replacement was made to replace obsolete valves whose normal maintenance components are no longer available. The new valves have threaded connections to facilitate removal for maintenance and testing. The replacement valves were determined to be equal or better than the removed valves.

Safety Evaluation:

The replacement valves were considered a one-for-one replacement and will not affect the plant safety or operation.

This modification does not have any adverse effect on the plant safety or operation. 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 87-004

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	08/27/90

QUALIFICATION (Q) LIST CHANGE PACKAGE

Summary:

The Total Equipment Data Base (TEDB) has been updated and approved as the new design database to replace the Q list at Turkey Point. The Q list was deleted and the TEDB will now be used as a source for design information. This PC/M involved no plant modifications or construction.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-090

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	08/03/90

REVISION OF ENVIRONMENTAL QUALIFICATION (EQ) DOCUMENTATION PACKAGES PER NRC AUDIT

Summary:

This Engineering Package issues revisions to the Environmental Qualification (EQ) list (5610-E-1435) and to the following EQ Documentation Packages: 5.0, 5.1, 6.0, 6.2, 11.0, 13.0, 15.0, 17.0, 17.1, 21.0, 21.1, 23.0, 25.0, 28.0, 34.2, 36.0, 1000, and 1001. This PC/M only updates the EQ Doc. Pacs. and EQ list and involves no plant modifications or construction.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-114-01

PC/M CLASSIFICATION : QR
UNIT : 3
TURN OVER DATE : 10/04/90

REPLACEMENT OF CCW HEAT EXCHANGER THERMOCOUPLES WITH RTDS

Summary:

This Engineering Package covers the replacement of the 12 locally installed Component Cooling Water (CCW) heat exchanger thermocouples with RTDs. Existing thermometers are not accurate enough to support the surveillance program and precision test instruments require approximately four hours of set-up and take down time for every hour of test time.

All RTDs will be installed in existing thermowells to avoid changes to the heat exchanger pressure boundaries. Two selector switches and a digital thermometer mounted in a nearby enclosure will be connected to the RTDs.

Although the monitoring equipment and modifications implemented by this package are not Safety Related, the enclosure and interconnecting conduits were seismically mounted to prevent potential damage to existing Safety Related equipment in the area.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-115-02

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	10/05/90

REPLACEMENT OF CCW HEAT EXCHANGER THERMOCOUPLES WITH RTDS

Summary:

This Engineering Package covers the replacement of the 12 locally installed CCW heat exchanger thermocouples with RTDs. Existing thermometers are not accurate enough to support the surveillance program and precision test instruments require approximately four hours of set-up and take down time for every hour of test time.

All RTDs will be installed in existing thermowells to avoid changes to the heat exchanger pressure boundaries. Two selector switches and a digital thermometer mounted in a nearby enclosure will be connected to the RTDs.

Although the monitoring equipment and modifications implemented by this package are not Safety Related, the enclosure and interconnecting conduits were seismically mounted to prevent potential damage to existing Safety Related equipment in the area.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-205

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 12/20/90

LOW PRESSURE TURBINE ROTOR REPLACEMENT

Summary:

This Engineering Package provides for the replacement of the Unit 4 low pressure turbine rotor. The change out of the turbine rotor is a like for like replacement of equipment.

The new turbine rotor replaced the older disc and shaft design with the latest available technology. The new rotor is a single machined extrusion which eliminates discs and keyways as well as other locations which are highly susceptible to stress corrosion cracking. In addition, crack growth is inhibited by the selection of materials which have lower ultimate strength and increased ductility.

To ensure that the Low Pressure Turbine Rotor will not catastrophically fail from stress corrosion cracking, the rotors should be inspected at the interval recommended by the Vendor. The vendor recommended interval is 40 years. As a good maintenance practice, it is FPL's intention to inspect the rotors at a 5 year interval.

Safety Evaluation:

There are no accidents or malfunctions of a different type from the safety analysis created by this modification. This modification does not have any adverse effect on the plant safety or operation. 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 87-316

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 08/27/90

Q LIST CHANGE PACKAGE

Summary:

This Engineering Package provides for the Q list change package. This PC/M only updates the Environmental Qualification (EQ) list and involves no plant modifications or construction.

The Total Equipment Data Base (TEDB) has been updated and approved as the new design database to replace the Q list at Turkey Point. The Q list was deleted and the TEDB will now be used as a source for design information. The changes in the above package have been incorporated in the TEDB.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-333-04

PC/M CLASSIFICATION	:	QR
UNIT	:	3/4
TURN OVER DATE	:	12/13/90

WASTE STORAGE FACILITY

Summary:

This Engineering Package provides for the addition of a new Waste Storage Facility and accompanying tie-ins such as water and fire protection.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-386-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	02/25/91

SECURITY BARRIERS FOR HVAC OPENINGS 56, 57, 58, AND 59

Summary:

This Engineering Package provides for the addition of new security barriers for HVAC openings 56, 57, 58, and 59. This package reinstalled magnets on new fire doors D067-1, D068-1, D070-1, D071-1, D094-1 and D096-1 and reinstalled and rewired conductive hinges and electric strikes on new fire doors D094-1 and D096-1.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 87-405-02

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 02/01/91

INSTALLATION OF HIGH DENSITY SPENT FUEL STORAGE RACKS

Summary:

This Engineering Package provides the necessary design, documentation, references, and instructions to remove, decontaminate, and replace the existing Unit 4 spent fuel storage racks with new vendor designed high density spent fuel storage racks.

The expansion of the spent fuel storage capacity was previously licensed for the vendor designed high density storage racks for both Units 3 and 4.

The new high density spent fuel storage racks are of similar material and design as the high density racks supplied for Unit 3. The Unit 4 high density racks were designed, fabricated, and installed by the same vendor used for the Unit 3 racks.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. This modification does not constitute an unreviewed safety question or require changes to the plant Technical Specifications. The Technical Specifications were previously amended to allow for spent fuel storage expansion from 621 spaces to 1404 spaces for each spent fuel pool. Therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 88-016-01

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	10/04/90

DETECTION SYSTEM FOR MONITORING REACTOR COOLANT SYSTEM LEAKS IN THE REACTOR HEAD AREA

Summary:

This Engineering Package provides for the installation of a reactor vessel head area leakage detection system. The detection system draws reactor head area or containment atmospheric samples into a skid mounted particulate sample system located inside containment. The system consists of a particulate detector, sample pump, motor operated valves, and various apparatus which interfaces with a remote control and display rack located in the control room. The remote rack consists of a four pen strip chart trend recorder, controller, digital ratemeter, and associated equipment. The system is powered from non-vital AC.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-078-02

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 08/29/90

RESIDUAL HEAT REMOVAL PUMPS MECHANICAL SEAL AND SEAL COOLER REPLACEMENT

Summary:

This Engineering Package provides for the replacement of the residual heat removal (RHR) pumps mechanical seal and seal cooler. The new seals are an upgraded design and have larger capacity seal coolers.

The removed seals had demonstrated an unsatisfactory seal life. The new seals are an upgraded design. The upgraded design seals coupled with the increased cooling capacity should provide a longer seal life.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-089

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	08/23/90

TURBINE STOP VALVES SAMPLE LINES

Summary:

This Engineering Package provides for the temporary installation and subsequent removal of monitoring instrumentation for the left turbine stop valve (4-010).

During plant operation in 1988, the left turbine stop valve closed without a signal to close. Subsequent testing could not cause the failure to repeat. Working with the vendor, maintenance determined the probable cause to be blockage of the control ports in the stop valve servo motor. The monitoring instrumentation was installed in an attempt to determine the cause of the event if it reoccurs.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-094-02

PC/M CLASSIFICATION : SR
UNIT : 3
TURN OVER DATE : 06/11/91

REDUCTION IN VOLTAGE DROP OF 4 KV BREAKER CONTROL CIRCUITS

Summary:

Previous engineering reviews performed under FPL DWA 60604 identified that the manufacturer's minimum voltage rating of 90 VDC for the closing circuit of certain safety related 4160V breakers could not be assured unless the vital 125 volt VDC system battery terminal voltage exceeded its 105 VDC minimum design basis value as stated in FSAR Section 8.2.3. Design modifications were performed by PC/M 83-154 on selected Unit 3 breakers to add an imposing relay which effectively reduced the voltage drop in the breaker closing circuit by reducing the circuit length from the associated battery terminals to the closing circuit. This assured breaker operation at a battery terminal voltage of 105 VDC. Safety Evaluation JPE-PTN-SELJ-88-030, Rev. 1, dated August 19, 1988, provided the basis for continued operation until the design modifications could be implemented. Based on actual breaker testing and discussions with the breaker manufacturer, it became apparent that breaker closing operation at 80 VDC could be verified, thus eliminating the need for the interposing relay. Therefore it was decided to remove the interposing relays added by PC/M 83-154 and rewire the closing circuits in accordance with the original design.

This Engineering Package provides for the removal of the interposing relays from the breaker closing circuit and provides the guidelines and requirements for testing the Unit 3, 4160 V safety related breakers to certify their operation at 80 VDC.

Safety Evaluation:

The design basis was reviewed to ensure that the overall design concept meets the applicable UFSAR, Regulatory Guides, and 10 CFR 50 requirements. The design analysis for the reduction in voltage drop of the 4KV breaker control circuits was verified by review of the calculations and the analytical techniques utilized. This modification does not have any adverse effect on the plant safety or operation. 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 88-095-02

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	06/28/91

REDUCTION IN VOLTAGE DROP OF 4 KV BREAKER CONTROL CIRCUITS

Summary:

Previous engineering reviews performed under FPL DWA 60604 identified that the manufacturer's minimum voltage rating of 90 VDC for the closing circuit of certain safety related 4160V breakers could not be assured unless the vital 125 volt VDC system battery terminal voltage exceeded its 105 VDC minimum design basis value as stated in FSAR Section 8.2.3. Design modifications were performed by PC/M 88-95 on selected Unit 4 breakers to add an imposing relay which effectively reduced the voltage drop in the breaker closing circuit by reducing the circuit length from the associated battery terminals to the closing circuit. This assured breaker operation at a battery terminal voltage of 105 VDC. Safety Evaluation JPE-PTN-SELJ-88-030, Rev. 1, dated August 19, 1988, provided the basis for continued operation until the design modifications could be implemented. Based on actual breaker testing and discussions with the breaker manufacturer, it became apparent that breaker closing operation at 80 VDC could be verified, thus eliminating the need for the interposing relay. Therefore it was decided to remove the interposing relays added by PC/M 88-95 and rewire the closing circuits in accordance with the original design.

This Engineering Package provides for the removal of the interposing relays from the breaker closing circuit and provides the guidelines and requirements for testing the Unit 4, 4160 V safety related breakers to certify their operation at 80 VDC.

Safety Evaluation:

The design basis was reviewed to ensure that the overall design concept meets the applicable UFSAR, Regulatory Guides, and 10 CFR 50 requirements. The design analysis for the reduction in voltage drop of the 4KV breaker control circuits was verified by review of the calculations and the analytical techniques utilized. This modification does not have any adverse effect on the plant safety or operation. 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 88-098

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 05/16/91

FEEDWATER HEATER BYPASS VALVE STROKE CLOSING TIME

Summary:

The Feedwater Heater Bypass Valve opens to provide condensate flow to the Steam Generator Feedwater 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 heaters. This could potentially trip the SGFPs which could result in tripping of the reactor. As documented in LER 84-21, a similar incident occurred in June, 1984, which caused a reactor trip.

To eliminate this problem, this engineering package provides modifications to the pneumatic controls of Feedwater Heater Bypass 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 closing, thereby lengthening the CV-2011 bypass valve closure time.

Safety Evaluation:

Bypass Valve CV-2011 does not perform any Safety Related function. However, the failure of the valve in the open position may cause the "Reduction in Feedwater Enthalpy" incident analyzed in the Updated Final Safety Analysis Report (UFSAR).

This modification does not have any adverse effect on the plant safety or operation. 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 88-143-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	07/03/90

ROADWAY AND SITE IMPROVEMENTS

Summary:

This Engineering Package provides for the addition of a new bridge over Lake Warren and the upgrading of the two intersections adjacent to the new bridge. The new bridge and roadway improvements do not perform any nuclear safety related functions.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-145

PC/M CLASSIFICATION : QR
UNIT : 3/4
TURN OVER DATE : 11/30/90

PERMANENT INSTALLATION OF DEMINERALIZED WATER STORAGE TANK (DWST) DRAIN HEADER

Summary:

This Engineering Package provides design documentation for the permanent installation of a new DWST drain header. The primary purpose of the header is to provide a multiple hookup point for supplying demineralized water to the DWST from the temporary demineralized water trailers. This header also provides an alternate means to fill the DWST.

This header replaced the temporary header installed under TSA 3-8-12-64.

Safety Evaluation:

The Technical Specifications require a minimum of 60,000 gallons (DWST level of five feet) to be maintained in the DWST to support the Standby Feedwater System availability in Modes 1, 2, and 3 as a non-safety related backup to the Auxiliary Feedwater System.

This modification does not have any adverse effect on the plant safety or operation. 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 88-149-01

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	07/25/90

RHR SYSTEM - REVISED OPERATION

Summary:

The Turkey Point Units 3 and 4 Emergency Core Cooling System (ECCS) and containment spray flows were evaluated for various system alignments during the post-LOCA injection and recirculation phases using a model developed in support of the FP&L Design Basis Reconstitution Program. Two scenarios were identified by the vendor which had the potential to result in insufficient NPSH for the Residual Heat Removal (RHR), Containment Spray (CS), and High Head Safety Injection (HHSI) pumps during the recirculation phase of LOCA recovery. These scenarios occur only during the safeguard pumps "piggy-back" (series) alignment mode, which is when the RHR pumps take suction from the containment sump and provide their discharge to the suction of the HHSI and/or CS pumps. One scenario consists of two CS and two HHSI pumps taking suction from one RHR pump. This scenario is prevented by administrative controls in the abnormal procedures. The second scenario with one CS and two HHSI pumps taking suction from one RHR pump, consisted of flashing occurring if valve 3-887 is throttled to a position of 30 percent open. The second scenario can be prevented by changing the normal valve line-up of valve 3-887 to 100 percent open in Modes 1-5.

This Engineering Package provides the required procedural and valve alignment changes covering operation of the Residual Heat Removal (RHR) system to preclude these two scenarios.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-150-01

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	07/25/90

RHR SYSTEM - REVISED OPERATION

Summary:

The Turkey Point Units 3 and 4 Emergency Core Cooling System (ECCS) and containment spray flows were evaluated for various system alignments during the post-LOCA injection and recirculation phases using a model developed in support of the FP&L Design Basis Reconstitution Program. Two scenarios were identified by the vendor which had the potential to result in insufficient NPSH for the Residual Heat Removal (RHR), Containment Spray (CS), and High Head Safety Injection (HHSI) pumps during the recirculation phase of LOCA recovery. These scenarios occur only during the safeguard pumps "piggy-back" (series) alignment mode, which is when the RHR pumps take suction from the containment sump and provide their discharge to the suction of the HHSI and/or CS pumps. One scenario consists of two CS and two HHSI pumps taking suction from one RHR pump. This scenario is prevented by administrative controls in the abnormal procedures. The second scenario with one CS and two HHSI pumps taking suction from one RHR pump, consisted of flashing occurring if valve 4-887 is throttled to a position of 30 percent open. The second scenario can be prevented by changing the normal valve line-up of valve 4-887 to 100 percent open in Modes 1-5.

This Engineering Package provides the required procedural and valve alignment changes covering operation of the Residual Heat Removal (RHR) system to preclude these two scenarios.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-168

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	06/06/91

REACTOR CAVITY HANDRAIL AND SAFETY CABLES

Summary:

This Engineering Package provides for the addition of safety handrails adjacent to the reactor cavity refueling pit at elevation 58 feet and for a safety cable system along the manipulator crane rails at elevation 58 feet.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-171

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	01/29/91

REPLACEMENT OF SPENT FUEL POOL BRIDGE CRANE

Summary:

This Engineering Package provided for the replacement of the Unit 4 Spent Fuel Pool Bridge Crane.

The new bridge crane is functionally similar to the old crane, with modifications to improve reliability, ease of operation, and maintenance. A fixed pointer position indicator system similar to that currently installed on the Unit 3 crane was installed on the new crane.

Safety Evaluation:

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-172

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	10/04/90

CONTAINMENT PURGE VALVE ACTUATOR VENTING MODIFICATION

Summary:

This Engineering Package provided for the replacement of the spring charging motor for the purge valve actuator with a new motor with the same mounting and electrical performance, but with a motor housing that is redesigned with ventilation holes.

Safety Evaluation:

The component design function is equivalent to or better than the original, providing the same or greater margins of safety as stated in the Technical Specifications. The component design intent is not changed with the equivalent replacement process. The functional description in the SAR is not changed. The motor has been changed to another model from a different manufacturer that meets or exceeds the original specifications or system design requirements.

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-249

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 03/06/91

REPLACEMENT OF RHR HEAT EXCHANGER OUTLET TO RWST VALVE 4-887

Summary:

This Engineering Package provides for the replacement of an 8 inch butterfly valve with an 8 inch globe valve on the Residual Heat Removal (RHR) heat exchanger outlet to the Reserve Water Storage Tank (RWST) valve 4-887. This modification was made due to a history of excessive leakage by the butterfly valve.

This modification provides a leak-tight isolation valve for required testing of the alternate low head Safety Injection flowpath. The replacement valve provides better flow throttling capability to the RWST during certain refueling operations.

Safety Evaluation:

It is noted that this valve modification provides replacement of an existing butterfly valve with a globe valve that meets or exceeds the original design requirements.

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-253

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	04/11/91

STEAM GENERATOR FEED RING J-NOZZLE REPLACEMENT

Summary:

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

Inspection of the steam generators revealed wall thinning in the carbon steel J-nozzles. Laboratory testing and wall thickness inspection data from operating steam generators were studied to obtain an improved J-nozzle design, featuring erosion/corrosion resistant materials.

This Engineering Package provides for the replacement of the existing carbon steel steam generator feed ring J-nozzles with new J-nozzles made from Inconel.

Safety Evaluation:

The steam generator feedring J-nozzles do not perform a Safety Related function, are not required to maintain reactor coolant system pressure boundary, and do not adversely impact plant safety or operations. However, the feed ring is seismically supported to preclude potential interactions with safety related equipment.

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-263-01

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 09/19/90

COMPONENT COOLING WATER HEAT EXCHANGER REPLACEMENT

Summary:

The Component Cooling Water System (CCW) serves as an intermediate barrier between the Intake Cooling Water (ICW) System and systems that are potential sources of radioactivity to reduce the probability of an uncontrolled radioactive material release. The CCW system, through the ICW system provides heat transfer capability to several engineering safety feature systems during normal plant operation, cooldown, and post accident conditions. These systems are required to function to achieve and maintain the plant in a safe shutdown condition.

This Engineering Package provides for the replacement of the CCW heat exchangers. Eddy current testing had revealed wall thinning and localized pitting of the existing heat exchanger tubes. In addition, visual inspection had revealed erosion of the heat exchanger tube sheets. Continued operation with the heat exchangers in the as found condition would reduce the heat transfer capability to engineered safety features and reduce the integrity of the systems's pressure boundary.

This modification is viewed as an interim solution to the corrosion and erosion problems experienced by the existing heat exchangers until such time as a complete failure analysis and permanent solution to the Intake Cooling Water (ICW) system problems are resolved.

The replacement units are similar to the existing units except for a higher rated flow rate without excessive vibration.

Safety Evaluation:

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-268

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	02/28/91

DIGITAL DATA PROCESSING SYSTEM GROUNDING

Summary:

This Engineering Package provided for the alteration of the Digital Data Processing System (DDPS) from an ungrounded system to a grounded system. Previously, the normal source (DDPS invertor output) was grounded, but the alternate sources were ungrounded. This modification added two grounded conductors, modified the ground detector system, deleted fuses on the grounded leg of the power source, and relabeled the switch boxes.

Safety Evaluation:

The DDPS is not required for the safe shutdown of the plant. The DDPS power feeds as modified by this PC/M are classified Non-Safety Related, performs no safety function and do not interface with safety related equipment. This evaluation has determined that this modification will have no adverse effect on the operation of any other system.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-284-01

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	08/23/90

UPGRADED CATHODIC PROTECTION SYSTEM AT CONDENSER WATERBOXES

Summary:

This Engineering Package provided for upgraded cathodic protection at the Unit 4 intake and discharge waterboxes. The previous cathodic protection system utilized carbon graphite anodes and manually controlled rectifiers. The upgraded system uses mixed metal oxide coated titanium tubular anodes and auto-potential controlled rectifiers. In addition, reference electrodes are provided at the tube-sheets. The reference electrodes allows the system engineer to obtain information on electrical potential gradients at various locations on the condenser intake and discharge tube-sheets.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 88-302-02

PC/M CLASSIFICATION : NNSR
UNIT : 4
TURN OVER DATE : 09/05/90

NUMBER 6 REHEATER EXTRACTION LINE REPLACEMENT

Summary:

In response to Information Notice 86-03, dealing with the problem of erosion/corrosion in high energy line piping, Turkey Point developed an ongoing inspection program for the secondary side main steam and feedwater piping. Based on the results of the latest inspection on Unit 3, the decision was made to replace the Unit 4 extraction lines from the high pressure turbine to the number 6 feedwater heaters.

This Engineering Package provides for the replacement of the existing 12 inch carbon steel piping with 12 inch chrome-moly piping in the number 6 reheater extraction line. This replacement was performed as the result of the latest Unit 4 piping inspection.

Safety Evaluation:

The new piping material has been demonstrated to have increased resistance to erosion/corrosion while the properties such as weight and tensile strength are comparable to those of the carbon steel piping removed. The extraction piping is physically located in the turbine building and has no safe shutdown function. The piping is non-seismically supported and is not within the seismic or Q piping boundaries identified for Turkey Point.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-304

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	03/12/91

MAIN CONTROL ROOM DOOR AND SECURITY SYSTEM MODIFICATIONS

Summary:

This Engineering Package provided for the replacement of the lockset on the main control room door. In addition, emergency lighting was installed in the control room foyer and a latchbolt guard plate was installed on the door's exterior to prevent forced entry.

These modifications are necessary to improve the door's reliability, to increase personnel safety, and to reduce the need for continued door maintenance.

Safety Evaluation:

The implementation of this modification does not change the functional or operational requirements of the existing system as required for normal plant safety. There is no adverse effect on plant operation or plant safety.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-394-01

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	10/04/90

INTAKE COOLING WATER OVERCURRENT TRIP

Summary:

There are three Intake Cooling Water (ICW) pumps available to meet the needs of the plant. Although the plant Technical Specifications call for all three pumps to be operable during plant operation, the plant is designed such that two pumps are sufficient for normal operations and one pump is sufficient for a Design Basis Accident (DBA).

ICW pump A is supplied from 4160V bus A while ICW pumps B and C are supplied from 4160V bus B. Under normal conditions with offsite power available, the buses can accommodate any combination of pumps. With a Loss of Offsite Power, the two 4160V buses are supplied from the A and B train emergency diesel generators, one specific to each bus. However, with the auto start logic of the standby ICW pump, there exists the potential to overload the B train emergency diesel generator under certain conditions.

To eliminate this potential for overloading an emergency diesel generator and to maintain consistency between the control circuits of the three ICW pumps, minor modifications were designed for the control circuits of all three pumps, to defeat the overcurrent trip interlock between the pumps. These modifications will not affect any other control functions in the starting circuits nor will they prevent a single pump from tripping due to overcurrent on any of its three phases. However, if an ICW pump trips on overcurrent, it will not cause an auto start of either of the two remaining pumps. It will be the operators decision to manually start an additional ICW pump, based on the requirements for cooling water and electrical loading of the emergency diesel generators.

Safety Evaluation:

The implementation of this modification does not change the functional or operational requirements of the existing system as required for normal plant safety. There is no adverse effect on plant operation or plant safety.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-462

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3
TURN OVER DATE	:	02/05/91

ELECTRICAL GENERATOR ROTOR - RADIAL LEAD MODIFICATION

Summary:

This Engineering Package brings the design of the electrical generator rotor radial leads up to the current design standard. This change is being made by the vendor as the result of an unacceptable pressure test on the Unit 3 rotor.

A hole in the end of the radial lead is being enlarged to permit a larger bore tightening tool to be used in the installation of the radial lead. This modification permits a more secure installation to be accomplished.

Safety Evaluation:

This modification does not affect the electrical characteristics of the device, but is being performed to upgrade the previous design. There is no change in the structural integrity of this part as the result of this modification.

The implementation of this modification does not change the functional or operational requirements of the existing system as required for normal plant safety. There is no adverse effect on plant operation or plant safety.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-468-01

PC/M CLASSIFICATION : SR
UNIT : 3/4
TURN OVER DATE : 04/03/91

ENVIRONMENTAL QUALIFICATION DOCUMENT PACKAGE REVISIONS

Summary:

Replacement of the accumulator pressure transmitters amplifier and calibration circuit boards must be performed every 10 years in accordance with maintenance note 16 on the 10 CFR 50.49 Environmental Qualification list.

These pressure transmitters were installed under PC/M 82-95. The EQ maintenance cycle for this activity must start from the date of installation of the transmitters.

Safety Evaluation:

The implementation of this modification does not change the functional or operational requirements of the existing system as required for normal plant safety. There is no adverse effect on plant operation or plant safety.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval.

PLANT CHANGE/MODIFICATION 88-488

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	12/19/90

CONCRETE REPAIR AT INTAKE

Summary:

This Engineering Package provides for the repair of concrete cracks and disbonded grout in the intake structure bays supporting the 4A1 circulating water pump, the screen-wash pumps, and the 4B ICW pump.

NCR C-0244-88 documented the existence of cracks in the faces of the concrete slabs. The NCR also documented the existence of cracked and disbonded grout and exposed rebar.

NCR C-0628-88 documented the existence of several pieces of rebar that were found to be damaged and corroded.

The affected concrete slabs were repaired to their original dimensions utilizing grouts specified and approved by project specification 5177-074-C-103. These grouts exhibit higher compressive strengths than the original concrete. In addition, the rebars, shims, and sole plate surfaces exposed during the repair process were coated with a corrosion resistant material to minimize any further corrosion.

Safety Evaluation:

The implementation of this modification does not change the functional or operational requirements of the existing system as required for normal plant safety. There is no adverse effect on plant operation or plant safety.

This modification was evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M did not require prior NRC approval. A Mode 4 restriction imposed by NCR C-0628-88 was lifted when this repair work was completed.

PLANT CHANGE/MODIFICATION 89-113

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	03/08/91

BORIC ACID BLEND FLOW CONVERTER RELOCATION AND REPLACEMENT AND SOLENOID VALVE REPLACEMENT

Summary:

This Engineering Package provides for the replacement and relocation of the boric acid flow converter (FM-4-113) and the replacement of solenoid valve SV-4-113A2.

Prior to this modification, the flow converter was located 494 feet away from the flow transmitter and was not powered from the same circuit as the transmitter. Signal interference due to the long transmission distance, powering of the transmitter and the converter from two different power sources prevented the proper performance of the converter. Furthermore, the existing flow converter was obsolete and needed to be replaced. In addition, solenoid valve SV-4-113A2 was obsolete and in need of replacement.

Safety Evaluation:

The implementation of this modification did not change the functional or operational requirements of the existing systems.

This modification does not have any adverse effect on the plant safety or operation. 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 89-168-01

PC/M CLASSIFICATION : SR
UNIT : 3
TURN OVER DATE : 06/28/91

ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) MODIFICATIONS

Summary:

An anticipated transient without a scram (ATWS) event is an anticipated operational transient which is accompanied by a failure of the reactor protection system (RPS) to shut down the reactor. Paragraph (c)(1) of 10 CFR 50.62 requires that all PWRs have a backup system to automatically initiate the auxiliary feedwater system and trip the turbine in case of an ATWS event. Florida Power and Light (FPL) prepared a conceptual design to provide such a system called ATWS mitigating system actuation circuitry (AMSAC).

The purpose of this modification is to meet the basic requirements of 10 CFR 50.62 by ensuring that a turbine trip occurs (to conserve steam inventory) and auxiliary feedwater is delivered to the steam generators to maintain an adequate secondary heat sink during a postulated ATWS event.

This Engineering Package provides design changes for the implementation of AMSAC for Unit 3.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. This modification does not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Normal plant administrative procedures are sufficient to control AMSAC. Technical Specifications are not required for AMSAC. Therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 89-169-01

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 06/28/91

ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) MODIFICATIONS

Summary:

An anticipated transient without a scram (ATWS) event is an anticipated operational transient which is accompanied by a failure of the reactor protection system (RPS) to shut down the reactor. Paragraph (c)(1) of 10 CFR 50.62 requires that all PWRs have a backup system to automatically initiate the auxiliary feedwater system and trip the turbine in case of an ATWS event. Florida Power and Light (FPL) prepared a conceptual design to provide such a system called ATWS mitigating system actuation circuitry (AMSAC).

The purpose of this modification is to meet the basic requirements of 10 CFR 50.62 by ensuring that a turbine trip occurs (to conserve steam inventory) and auxiliary feedwater is delivered to the steam generators to maintain an adequate secondary heat sink during a postulated ATWS event.

This Engineering Package provides design changes for the implementation of AMSAC for Unit 4.

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. This modification does not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Normal plant administrative procedures are sufficient to control AMSAC. Technical Specifications are not required for AMSAC. Therefore, prior NRC approval was not required for this modification.

PLANT CHANGE/MODIFICATION 89-209

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	08/31/90

QUICK DISCONNECTS FOR SUCTION STABILIZER VENT VALVES

Summary:

The previous method of venting the charging pump suction stabilizers involved contacting maintenance for the installation of a hose from the vent valve to the radwaste floor drain. This required the approval of a work order, which could take eight hours. The installation of the hose, venting of the stabilizer, and removal of the hose, can take an additional four hours. During plant operation, each stabilizer requires daily venting.

This Engineering Package provides a means to simplify the venting of the charging pump stabilizers. The installation of quick disconnects on the vent valves will allow an operator to vent the stabilizers without a work order and without assistance from maintenance.

Safety Evaluation:

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 89-210

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 09/04/90

QUICK DISCONNECTS FOR SUCTION STABILIZER VENT VALVES

Summary:

The previous method of venting the charging pump suction stabilizers involved contacting maintenance for the installation of a hose from the vent valve to the radwaste floor drain. This required the approval of a work order, which could take eight hours. The installation of the hose, venting of the stabilizer, and removal of the hose, can take an additional four hours. During plant operation, each stabilizer requires daily venting.

This Engineering Package provides a means to simplify the venting of the charging pump stabilizers. The installation of quick disconnects on the vent valves will allow an operator to vent the stabilizers without a work order and without assistance from maintenance.

Safety Evaluation:

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 89-274

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	03/28/91

ENVIRONMENTAL QUALIFICATION (EQ) DOCUMENT PACKAGE REVISIONS - EQ AUDIT FOLLOW-UP

Summary:

This Engineering Package provides the mechanism for revising Environmental Qualification Documentation Packages (EQ Doc Pacs). These revisions are required to address NRC comments from the December, 1988 EQ Audit. This PC/M requires no physical changes to the plant.

The EQ list for 10 CFR 50.49 was revised to reflect the changes required by this engineering package.

The Environmental Qualification Document Packages being revised are as follows:

Conax Electrical Penetrations
Okonite Cables
Target Rock Solenoid Valves
Valcor Solenoid Valves
Generic Approach and Treatment of Issues

Safety Evaluation:

This modification does not have any adverse effect on the plant safety or operation. 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 89-326

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	07/25/90

REPLACEMENT OF RHR HEAT EXCHANGER OUTLET TO RWST VALVE 3-887

Summary:

This Engineering Package provides for the replacement of an 8 inch butterfly valve with an 8 inch globe valve on the Residual Heat Removal (RHR) heat exchanger outlet to the Reserve Water Storage Tank (RWST) valve 3-887. This modification was made due to a history of excessive leakage by the butterfly valve.

This modification provides a leak-tight isolation valve for required testing of the alternate low head Safety Injection flowpath. The replacement valve provides better flow throttling capability to the RWST during certain refueling operations.

Safety Evaluation:

It is noted that this valve modification provides replacement of an existing butterfly valve with a globe valve that meets or exceeds the original design requirements.

This modification has been evaluated in accordance with 10 CFR 50.59, and found not to give rise to any unreviewed safety questions, and to not affect the plant Technical Specifications. Consequently, implementation of this PC/M does not require prior NRC approval.

PLANT CHANGE/MODIFICATION 89-335-01

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	09/11/90

CONTAINMENT ELECTRICAL PENETRATION REPLACEMENT AND INSTALLATION

Summary:

This Engineering Package provided for the addition of new spare containment electrical penetration assemblies. These new assemblies are being installed to replace the spare conductors that were used as a result of the extensive modifications made for Appendix R requirements.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification will not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-379-01

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	03/28/91

MODIFICATION OF INSTRUMENT LOOPS IN RESPONSE TO NCR N-89-0709

Summary:

This Engineering Package provided for the modification of certain instrument loops to minimize common mode voltage exceeding the recommended limit of +/_ 10 volts to ground potential.

Non-Conformance Report N-89-0709 identified instrument 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 attributed with causing excessive loading on the primary instrument loop by a common mode voltage problem. The loops being modified are as follows:

LT-4-470	Pressurizer Relief Tank Level
LT-4-115	Volume Control Tank Level
LT-4-112	Volume Control Tank Level
FT-4-6274	Blowdown Heat Exchanger Outlet Flow
FT-4-6277A	Steam Generator A Blowdown Effluent Flow
FT-4-6277B	Steam Generator B Blowdown Effluent Flow
FT-4-6277C	Steam Generator C Blowdown Effluent Flow

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification does not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-420

PC/M CLASSIFICATION	:	SR
UNIT	:	3/4
TURN OVER DATE	:	03/20/91

RACEWAY PROTECTION FOR APPENDIX R

Summary:

This Engineering Package provided for the modification of supports, raceways, etc. to allow installation of thermo-lag fire wrap to meet Appendix R requirements. This PC/M also installed and in some cases removed thermolag fire wrap per Appendix R requirements.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification will not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval is not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-500

PC/M CLASSIFICATION	:	QR
UNIT	:	3
TURN OVER DATE	:	07/18/90

SI ACCUMULATORS LEVEL TRANSMITTER ACCESS PLATFORMS

Summary:

The Safety Injection (SI) Accumulator Tanks Level Transmitters (LT) were replaced and relocated via PC/M 88-461 in order to provide transmitters with a new range. This Engineering Package provides for the installation of permanent access platforms and lighting for maintenance and calibration of the safety injection accumulator tanks level transmitters in their new locations.

Safety Evaluation:

The level transmitter access platforms and lighting do not perform any safety related functions. However, they are associated with or located in the vicinity of safety related systems. Therefore, light fixtures and raceways will be seismically supported and the platforms will be seismically designed and installed to preclude interaction with other safety related systems, structures, or components. This modification is classified safety related since rework/rerouting of the SI Accumulator Level Transmitter instrument tubing and cable is required.

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification will not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-505

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	05/21/91

AUXILIARY FEEDWATER DC MOV SHUNT FIELD SURGE SUPPRESSION

Summary:

This Engineering Package modifies the 125 VDC motor operators for Auxiliary Feedwater motor operated valves, MOV-3-1403 and 1405, by adding shunt surge suppression resistors connected in parallel with the shunt field of the motor. This modification suppresses the energy buildup in the shunt field caused by self induction created when the motor is disconnected from its DC power source. This concern was addressed in NRC Information Notice 88-72 dated September 2, 1988.

These two MOVs serve as two of the three steam admission valves to provide steam to the auxiliary feedwater pumps upon loss of normal feedwater. They also provide isolation of the steam lines from the steam generators following a steam generator tube rupture.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification does not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-506

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	03/14/91

AUXILIARY FEEDWATER DC MOV SHUNT FIELD SURGE SUPPRESSION

Summary:

This Engineering Package modifies the 125 VDC motor operators for Auxiliary Feedwater motor operated valves, MOV-4-1403 and 1405, by adding shunt surge suppression resistors connected in parallel with the shunt field of the motor. This modification suppresses the energy buildup in the shunt field caused by self induction created when the motor is disconnected from its DC power source. This concern was addressed in NRC Information Notice 88-72 dated September 2, 1988.

These two MOVs serve as two of the three steam admission valves to provide steam to the auxiliary feedwater pumps upon loss of normal feedwater. They also provide isolation of the steam lines from the steam generators following a steam generator tube rupture.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification does not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-568

PC/M CLASSIFICATION : QR
UNIT : 4
TURN OVER DATE : 05/21/91

LOSS OF DECAY HEAT REMOVAL PROGRAMMED ENHANCEMENT RESIDUAL HEAT REMOVAL SYSTEM

Summary:

This Engineering Package provided for the modification of the Residual Heat Removal System flow indication by including an adjustable flow alarm in addition to the current fixed low flow alarm. Additional equipment includes the addition of a digital bar-graph indicator and a dual comparator module in place of the current single comparator module. Further, the control room annunciator window was engraved to reflect the new alarm conditions.

This modification was made to meet commitments made in FPL letter L-89-37 in response to NRC Generic Letter 88-17.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification does not have any adverse effects on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 89-574

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	08/22/90

REACTOR COOLANT PUMP 3C MOTOR REFURBISHMENT/UPGRADE

Summary:

As part of the on-going program to improve reliability and performance, Reactor Coolant Pumps (RCP) at Turkey Point are being factory refurbishment on a rotating basis. The factory refurbishment consisted of inspection and maintenance activities performed to factory specifications. In addition, two upgrade modifications were performed concurrent with the refurbishment of this motor. These refurbishments ensure consistency with the latest RCP technology and a greater RCP reliability.

This Engineering Package mainly addresses the refurbishment and upgrade modifications relative to the rotated spare motor. These modifications are nearly identical with the upgrade of RCP 4B (PC/M #88-450).

Safety Evaluation:

The RCP motor does not perform any safety related function with the exception of providing sufficient inertia (through its flywheel) to ensure sufficient coastdown of the RCP after a RCP or reactor trip.

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification will not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-027

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	03/27/91

UNIT 3 SAFETY INJECTION BLOCK SWITCH REPLACEMENT

Summary:

This Engineering Package provided for the replacement of the existing safety injection (SI) block/unblock switch with two physically independent selector switches, each dedicated to a single SI actuation logic train.

The removed switch was used to block the Safety Injection (SI) signal due to any of the following: 1) low pressurizer pressure; 2) high steam header/steam generator differential pressure; 3) high steam generator flow coincident with either low steam generator pressure or low $T_{avg.}$. The removed switch assembly simultaneously operated two independent safety related trains of logic.

Westinghouse alerted FPL via letter in 1989 that a single failure could compromise both SI actuation logic trains. Westinghouse, after their analysis of the single switch arrangement concluded that although it was not an immediate safety concern, they recommended that the design change be developed immediately to provide two independent switches. Immediate concerns were handled by procedural controls.

Safety Evaluation:

This evaluation has shown that this modification does not require a change to the Technical Specifications nor does it constitute an unreviewed safety question. This modification does not have any adverse effect on plant safety, security or operations. Therefore prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-232

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	05/30/91

MODIFICATION TO INCREASE CAPACITY OF RCP OIL COLLECTION TANK

Summary:

This Engineering Package provided for the modification of the Reactor Coolant Pump (RCP) oil collection tank by extending one end of the existing tank to accommodate the increased oil inventory of the "original spare" RCP motor. This tank provides means of collecting the entire lube oil inventory from one RCP motor and the expected normal leakage from the other two motors for a fuel cycle. This tank was designed to meet 10 CFR Appendix R requirements.

Safety Evaluation:

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-320

PC/M CLASSIFICATION : SR
UNIT : 4
TURN OVER DATE : 03/27/91

UNIT 4 SAFETY INJECTION BLOCK SWITCH REPLACEMENT

Summary:

This Engineering Package provided for the replacement of the existing safety injection (SI) block/unblock switch with two physically independent selector switches, each dedicated to a single SI actuation logic train.

The removed switch was used to block the Safety Injection (SI) signal due to any of the following: 1) low pressurizer pressure; 2) high steam header/steam generator differential pressure; 3) high steam generator flow coincident with either low steam generator pressure or low $T_{avg.}$. The removed switch assembly simultaneously operated two independent safety related trains of logic.

Westinghouse alerted FPL via letter in 1989 that a single failure could compromise both SI actuation logic trains. Westinghouse, after their analysis of the single switch arrangement concluded that although it was not an immediate safety concern, they recommended that the design change be developed immediately to provide two independent switches. Immediate concerns were handled by procedural controls.

Safety Evaluation:

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-338-01

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	05/24/91

POWER FOR BLACKSTART DIESEL GENERATORS BATTERY CHARGER

Summary:

This Engineering Package provides the implementation details to provide the Blackstart diesel generators battery charger with a back-up power source. An automatic bus transfer switch was added to automatically transfer the battery charger to the alternate power source if the normal power source is not available. In addition, a transformer was added to raise the back-up power voltage from 208 volts to 240 volts to eliminate the need to adjust the battery charger.

Safety Evaluation:

This modification does not affect the operation of the battery charger or any other circuitry for the Blackstart diesel generators. This modification has no potential seismic interaction concerns since the installation is on the fossil side of the plant site and not in the vicinity of any safety related equipment. In addition, the equipment being installed is classified Not Nuclear Safety Related.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-389

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	12/21/90

SPENT FUEL PUMPS - ISOLATION VALVE INSTALLATION

Summary:

This Engineering Package provides for the modification of the spent fuel pool (SFP) pump suction piping by increasing the diameter from 8 inches to 10 inches and adding an isolation valve. These modifications were made in anticipation of the future replacement of the current pump with a pump from a different manufacturer.

Implementation of the above modification will require temporary isolation of the flow to the Spent Fuel Pool. As described in UFSAR Section 9.3, SFP cooling may be safely shutdown for a reasonable time period for maintenance or replacement of malfunctioning components.

Safety Evaluation:

The Spent Fuel Pool Cooling System is not required to maintain reactor coolant system pressure boundary integrity or to assure capability to achieve or maintain safe shutdown, or to mitigate the consequences of accidents with potential offsite exposures approaching the 10 CFR 100 limits.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-401-01

PC/M CLASSIFICATION : QR
UNIT : 3
TURN OVER DATE : 01/04/91

INTAKE COOLING WATER VALVE PIT RIGGING BEAM

Summary:

This Engineering Package provides for the permanent installation of the intake cooling water (ICW) valve pit rigging beams for the ICW "Crawl Through" inspection/repair. This installation will eliminate the potential for accidental damage during removal and installation of temporary rigging beams. This evaluation addresses the permanent installation of the ICW valve pit rigging beams only. Activities associated with the use of these beams for rigging shall be considered separately in appropriate evaluation documents.

Safety Evaluation:

The rigging beams do not perform a safety function. These beams are attached to the overhead structural steel beams above the safety related ICW system. These beams are seismically designed to prevent any adverse interaction with safety related equipment.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-412-01

PC/M CLASSIFICATION	:	QR
UNIT	:	4
TURN OVER DATE	:	12/26/90

INTAKE COOLING WATER VALVE PIT RIGGING BEAM

Summary:

This Engineering Package provides for the permanent installation of the intake cooling water (ICW) valve pit rigging beams for the ICW "Crawl Through" inspection/repair. This installation will eliminate the potential for accidental damage during removal and installation of temporary rigging beams. This evaluation addresses the permanent installation of the ICW valve pit rigging beams only. Activities associated with the use of these beams for rigging shall be considered separately in appropriate evaluation documents.

Safety Evaluation:

The rigging beams do not perform a safety function. These beams are attached to the overhead structural steel beams above the safety related ICW system. These beams are seismically designed to prevent any adverse interaction with safety related equipment.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-413

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3
TURN OVER DATE	:	04/25/91

REMOVAL OF CONDENSATE PUMP RECIRCULATION INSTRUMENTATION AND REPLACEMENT OF OBSOLETE FLOW INSTRUMENTATION

Summary:

This Engineering Package provides the justification and instructions for the removal of the condensate pump recirculation instrumentation. Additionally, justification is provided for the isolation of the subject recirculation lines, as well as instructions for replacing the condensate pump discharge flow transmitters, flow switches, flow elements, and alarm relays. This instrumentation was removed because it was obsolete and unused. The recirculation lines have been valved out for the operating history of the plant.

Safety Evaluation:

This engineering package is classified as Not Safety Related because the instrumentation being removed or replaced and the valves being affected are classified as Not Safety Related. The condensate pumps recirculation piping and associated instrumentation perform no safety function.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-414

PC/M CLASSIFICATION	:	NNSR
UNIT	:	4
TURN OVER DATE	:	04/25/91

REMOVAL OF CONDENSATE PUMP RECIRCULATION INSTRUMENTATION AND REPLACEMENT OF OBSOLETE FLOW INSTRUMENTATION

Summary:

This Engineering Package provides the justification and instructions for the removal of the condensate pump recirculation instrumentation. Additionally, justification is provided for the isolation of the subject recirculation lines, as well as instructions for replacing the condensate pump discharge flow transmitters, flow switches, flow elements, and alarm relays. This instrumentation was removed because it was obsolete and unused. The recirculation lines have been valved out for the operating history of the plant.

Safety Evaluation:

This engineering package is classified as Not Safety Related because the instrumentation being removed or replaced and the valves being affected are classified as Not Safety Related. The condensate pumps recirculation piping and associated instrumentation perform no safety function.

This evaluation has shown that this modification does not have an adverse effect on plant safety or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-481

PC/M CLASSIFICATION	:	NNSR
UNIT	:	3/4
TURN OVER DATE	:	02/05/91

ADDITION OF BACK-UP LEVEL INDICATION TO THE ACID AND CAUSTIC STORAGE TANKS IN THE WATER TREATMENT PLANT

Summary:

This Engineering Package provides the justification and instructions for the addition of back-up local level indicators to the Water Treatment Plant (WTP) acid and caustic storage tanks. The additional level indicators will allow operators to verify actual tank levels while transferring chemicals to or from the subject tanks.

Safety Evaluation:

This engineering package was classified non safety related because the instrumentation being added and the tanks being affected are classified as not safety related. The water treatment system has no safety related design bases and performs no safety functions. The implementation of this modification does not change the functional or operational requirements of the existing system.

This evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does not require a change to the plant Technical Specifications. Thus prior NRC approval was not required for implementation of this modification.

PLANT CHANGE/MODIFICATION 90-524

PC/M CLASSIFICATION	:	SR
UNIT	:	3
TURN OVER DATE	:	06/14/91

SAFETY INJECTION ACCUMULATOR WATER LEVEL INSTRUMENT BAND INCREASE

Summary:

This Engineering Package provides the change of the range of the six accumulator level instrument channels to encompass the proposed Technical Specifications limits. This change was made concurrently with the application for a change to the plant Technical Specifications pertaining to the required quantity of borated water contained in the Safety Injection Accumulators.

The current operating band corresponds to approximately a one inch change in tank level. This narrow band causes frequent filling/venting operations and subsequent chemical analysis. The new Technical Specifications limits will allow an operating band of approximately five inches.

Safety Evaluation:

An administrative requirement prevents the unit from entering Mode 4 until this licensee amendment has been issued by the NRC. With this restriction in mind, this evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does requires a change to the plant Technical Specifications prior to entering Mode 3 from Mode 4 after implementation of the modification. Thus prior NRC approval was not required for implementation of this modification with this Mode restriction.

PLANT CHANGE/MODIFICATION 90-525

PC/M CLASSIFICATION	:	SR
UNIT	:	4
TURN OVER DATE	:	06/14/91

SAFETY INJECTION ACCUMULATOR WATER LEVEL INSTRUMENT BAND INCREASE

Summary:

This Engineering Package provides the change of the range of the six accumulator level instrument channels to encompass the proposed Technical Specifications limits. This change was made concurrently with the application for a change to the plant Technical Specifications pertaining to the required quantity of borated water contained in the Safety Injection Accumulators.

The current operating band corresponds to approximately a one inch change in tank level. This narrow band causes frequent filling/venting operations and subsequent chemical analysis. The new Technical Specifications limits will allow an operating band of approximately five inches.

Safety Evaluation:

An administrative requirement prevents the unit from entering Mode 4 until this licensee amendment has been issued by the NRC. With this restriction in mind, this evaluation has shown that this modification does not have an adverse effect on plant safety, security, or operation, or constitute an unreviewed safety question. This modification does requires a change to the plant Technical Specifications prior to entering Mode 3 from Mode 4 after implementation of the modification. Thus prior NRC approval was not required for implementation of this modification with this Mode restriction.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION: JPE-LR-87-020 Revision 4

OPERATION WITH LOSS OF HVAC TO THE DC EQUIPMENT AND INVERTER ROOMS

In 1986, Florida Power and Light (FPL) identified a concern related to the loss of HVAC in the DC Equipment/Inverter Rooms at Turkey Point Units 3 & 4.

This evaluation indicates a maximum operability temperature of 135 degrees Fahrenheit is acceptable (except for the battery rooms where 115 degrees Fahrenheit is the acceptable maximum). Operation at or below this temperature for short periods of time (from a few days to months depending on the particular piece of equipment) does not introduce a failure mechanism. This evaluation also reviewed the use of supplemental cooling and provided requirements that should be implemented to ensure detection and timely compensatory actions for any credible HVAC failure scenario and resulting temperature excursion in the DC Equipment/Inverter rooms.

Safety Evaluation Summary:

The DC Equipment/Inverter Rooms' HVAC systems are not addressed in the Turkey Point Technical Specifications; therefore, this evaluation has no effect on the plant's Technical Specifications. Since no existing safety analyses are affected and no new failure modes are introduced, the use of supplemental cooling does not constitute an unreviewed safety question pursuant to 10 CFR 50.59. Therefore, prior NRC approval was not required.

Revision 4 of this safety evaluation provides an evaluation of selected single-failure type loss of HVAC scenarios, for the post 1990-1991 dual unit outage configuration, to justify the acceptability of the modified design.

Issued: May 23, 1991

SAFETY EVALUATION: JPN-PTN-SEMJ-89-043 Revision 0

SAFE SHUTDOWN CAPABILITY WITH SPURIOUS CLOSURE OF VOLUME CONTROL TANK LEVEL CONTROL VALVE LCV-115C

The Appendix R Safe Shutdown Analysis takes credit for operator action to mitigate the adverse effects of spurious closure of Volume Control Tank Level Control Valve LCV-115C. However, without administrative controls or a permanent design change, this operator action may not be taken in time to prevent pump damage.

Safety Evaluation Summary:

This evaluation has shown that the implementation of administrative controls in the form of fire watches will ensure the availability of a charging pump for safe shutdown capability in the event of a fire and, therefore, does not result in an unreviewed safety question or require any changes to the plant Technical Specifications. Therefore, prior NRC approval for implementation of these temporary administrative controls was not required.

Issued: May 24, 1990

SAFETY EVALUATION: JPN-PTN-SENJ-89-048 Revision 3

EMERGENCY POWER SYSTEM (EPS) ENHANCEMENT REPORT, SUPPLEMENT NO. 2

Florida Power and Light (FPL) is installing two new emergency diesel generators and associated electrical and mechanical equipment at Turkey Point Nuclear. Supplement 1 submitted via FPL letter L-89-124 dated April 3, 1989, provided information regarding testing to be performed on various components and systems during turnover, startup, pre-operational testing, and prior to returning the enhanced EPS to service.

Supplement 2 provides an evaluation for the enhanced emergency power system.

Safety Evaluation Summary:

The information presented in this safety evaluation demonstrates that the enhanced EPS provides additional installed capacity at Turkey Point such that the design basis accident of Loss of Offsite Power, plus a Loss of Coolant Accident on one unit, plus the single failure of an EDG, is mitigated with 3 EDGs available. The three EDGs can be automatically loaded and manually loaded with the required loads for accident mitigation on one unit and the achievement of safe shutdown on the non-accident unit.

The Failure Modes and Effects Analysis, performed for the enhanced design, demonstrated that the minimum equipment to mitigate the design basis accidents described in the FSAR is readily available with the enhanced EPS configuration, even assuming a single failure of the EDG to start. Thus the accident analysis in the FSAR remains valid as the bounding analysis and the accident analysis results are not affected as a result of re-configuring the EPS by this enhancement project.

Issued: May, 1990

SAFETY EVALUATION: JPN-PTN-SEMJ-89-057 Revision 0

MODIFICATIONS TO FIRE DOORS

This safety evaluation analyzes a variety of minor modifications to fire-rated doors without invalidating the intended design function.

Safety Evaluation Summary:

This evaluation has shown that the types of modifications to the fire doors discussed within this safety evaluation are acceptable and did not result in an unreviewed safety question or require any changes to the plant Technical Specifications. Therefore, prior NRC approval for implementation of the modification to the fire door are not required. In addition, these modifications to the fire doors do not adversely affect plant operation and safety while ensuring compliance with Appendix R safe shutdown capability.

Issued: May 24, 1990

SAFETY EVALUATION: JPN-PTN-SEMJ-89-067 Revision 0

CHANGE TO ADMINISTRATIVE TEMPERATURE LIMIT ON RCS HEATUP AND COOLDOWN RATES

During a review of the FSAR, a discrepancy was noted in Section 4.2.6. The purpose of this safety evaluation was to evaluate the recommended change to the FSAR with regard to the heatup and cooldown rates for the reactor coolant system (RCS).

Safety Evaluation Summary:

The heatup and cooldown curves of the Technical Specifications were not affected by this revision of the FSAR.

This evaluation has shown that the FSAR revision does not have any adverse effect on plant operation or safety and that no unreviewed safety question or technical specification change were involved. The margin of safety as defined in the basis for any Technical Specification was not reduced since the RCS heatup and cooldown rates are governed by the curves in the plant Technical Specifications. Therefore, prior NRC approval for implementation was not required.

Issued: October 10, 1989

SAFETY EVALUATION: JPN-PTN-SEMJ-89-105 Revision 0

APPENDIX R - SAFETY EVALUATION FOR THE SAFE SHUTDOWN ANALYSIS AND ESSENTIAL EQUIPMENT LIST

This safety evaluation provides a basis for official plant approval of the Safe Shutdown Analysis (SSA) as a formal plant drawing.

Safety Evaluation Summary:

This evaluation has shown that the issuance of this drawing does not result in any plant modifications or changes to the design basis of the plant, but serves only to list specifically the minimum equipment which is used to attain and maintain safe shutdown in the event of a fire. This drawing does not involve any unreviewed safety questions or a change to the plant Technical Specifications, pursuant to 10 CFR 50.59. In addition, these changes do not adversely affect plant operation or safety. Therefore prior NRC approval was not required to implement these changes.

Issued: May 24, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-89-107 Revision 0

TURKEY POINT UNIT 4 TURBINE STOP VALVE EVALUATION

During a stroke test of the turbine stop valves in Mode 2 with full steam pressure but no steam flow, the turbine stop valves failed to fully close when stroked. This safety evaluation was prepared to assess the significance of a similar failure in the future.

<u>Safety Evaluation Summary:</u>

This evaluation has shown that the plant FSAR licensing basis accident analysis bounds all postulated plant transients that could occur due to the turbine stop valves not closing completely during the specific conditions described in the evaluation.

Issued: October 6, 1989

SAFETY EVALUATION: JPN-PTN-SENJ-89-130 Revision 4

WESTINGHOUSE OT-2 SWITCHES

Westinghouse informed Florida Power and Light by a letter dated October 26, 1989, of a potential single failure deficiency associated with the Control Room Safety Injection Block Switches.

Revision 1 incorporated additional information which address the potential for latent switch failures which may affect the use of the EOPs.

Revision 2 addressed additional switches identified by Westinghouse and provides overall clarification.

Revision 3 clarified and expounded on the evaluation for the feedwater isolation safeguards function on high steam generator level.

Revision 4 added an engineering review of all remaining safety related and quality related switches not previously analyzed in prior revisions of this evaluation.

<u>Safety Evaluation Summary:</u>

The above evaluation identified a potential single failure with the control room Safety Injection block switch and/or Containment Spray reset switch. Procedural changes to mitigate the potential for the existence of the postulated failure have been determined to not involve an unreviewed safety question and require no changes to the Technical Specifications. Therefore, prior NRC approval was not required.

Issued: December 21, 1990

SAFETY EVALUATION: JPE-PTN-SEIJ-89-139 Revision 1

RELOCATION OF ND 6700 CHEMISTRY ANALYZER TO COMPUTER ROOM

Chemistry Analyzer ND 6700 was designed to be installed in the Computer Room under PC/M 81-034. However, this PC/M was not fully implemented. Although all other construction was completed, this analyzer was not physically installed in the computer room at the request of the Chemistry Department. During the interim, the analyzer was located in the chemistry Hot Lab. Due to the Hot Lab environment, it experienced breakdowns and required frequent maintenance. The proposed change for eliminating the environmental problems was to relocate the analyzer to the computer room as originally designed under PC/M 81-034.

The analyzer continues to perform its intended function in its new location. Therefore, the relocation of the analyzer had no effect on plant operation or safety, and does not create any changes in the existing operating practices.

Safety Evaluation Summary:

The relocation of the analyzer did not change the function or operational requirements of the analyzer. There are no effects on plant operation or plant safety and relocation of the analyzer does not constitute an unreviewed safety question nor does it require changes to the plant Technical Specifications pursuant to 10 CFR 50.59. Therefore, prior NRC approval was not required for implementation of this request.

Issued: September 6, 1990

SAFETY EVALUATION: JNO-90-001 Revision 0

CHANGE IN COMPONENT COOLING WATER CHLORIDE SPECIFICATION SAFETY EVALUATION

Chemistry requested an evaluation of the change in the CCW chloride specification from 0.150 ppm to 1.0 ppm. The purpose of this change is to permit the corrosion inhibitor chemicals to be maintained in the range of 200 to 500 ppm to provide maximum protection of the CCW system materials while maintaining a safe margin to prevent chloride induced stress corrosion cracking (SCC). A site procedure has been written that allows control of the chloride concentration below 1.0 ppm by feed and bleed operations. The CCW system chemical control program uses a molybdate inhibitor which effectively protects austenitic stainless steel from SCC in chloride concentrations being recommended.

<u>Safety Evaluation Summary:</u>

Neither the plant Technical Specifications nor the plant updated FSAR specifies chemistry parameters to be maintained in the CCW system. Therefore this change does not require any changes to the Plant Technical Specifications.

Based on the performed evaluation, it is concluded that the change in the CCW system chloride specification is safe to perform and the change can be implemented without prior NRC approval because the change does not affect plant safety or operation pursuant to 10 CFR 50.59.

Issued: September 20, 1990

SAFETY EVALUATION: JPE-PTN-SENJ-90-011 Revision 0

COMPONENT COOLING WATER THERMAL BARRIER RELIEF VALVES

The NRC performed a Safety System Functional Inspection at Turkey Point in August, 1985. As a result of this inspection, it was concluded that insufficient design basis information was available for making engineering decisions. In response to this, Design Basis Documents (DBD) were prepared for support and accident mitigation systems, as well as selected licensing issues. As part of the DBD program for Turkey Point Units 3 & 4, a series of reviews were performed comparing the information provided in the DBDs with that contained in the FSAR. This review revealed an inconsistency between the FSAR and the DBD in the discussion of the function of the CCW relief valves RV-722-A, -B, and -C.

Safety Evaluation Summary:

This evaluation showed that the DBD description was correct and that the FSAR should be revised to reflect this change. This change to the FSAR does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 nor does it require a change to the Technical Specifications. Therefore, prior NRC approval was not required.

Issued: March 30, 1990

SAFETY EVALUATION: JPE-PTN-SENJ-90-012 Revision 0

COMPONENT COOLING WATER PASSIVE FAILURE

The NRC performed a Safety System Functional Inspection on the Turkey Point Plant in August, 1985. As a result of this inspection, it was concluded that insufficient design basis information was available for making engineering decisions. In response to this, Design Basis Documents (DBD) were prepared for support and accident mitigation systems, as well as Selected Licensing issues. As part of the DBD program for Turkey Point Units 3 & 4, a series of reviews were performed comparing the information provided in the DBDs with that contained in the FSAR. This review revealed an inconsistency between the FSAR and the DBD in the discussion of the function of the CCW header isolation valves.

Safety Evaluation Summary:

This evaluation shows that the DBD description was correct and that the FSAR should be revised to reflect this change. This change to the FSAR does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 nor does it require a change to the Technical Specifications. Therefore, prior NRC approval was not required.

Issued: March 30, 1990

SAFETY EVALUATION: JPE-PTN-SENJ-90-016 Revision 1

CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) DUAL PUMP OPERATION

The NRC performed a Safety System Functional Inspection on the Turkey Point Plant in August, 1985. As a result of this inspection, it was concluded that insufficient design basis information was available for making engineering decisions. In response to this, Design Basis Documents (DBD) were prepared for support and accident mitigation systems, as well as Selected Licensing issues. As part of the DBD program for Turkey Point Units 3 & 4, a series of reviews were performed comparing the information provided in the DBDs with that contained in the final safety analysis report (FSAR). This review revealed an inconsistency between the FSAR and the DBD in the discussion of the CVCS. The FSAR used the phrase "time periods are halved." The DBD used the phrase "reduce the time, but will not halve the time."

Safety Evaluation Summary:

This evaluation shows that the change of "halved" to "reduced" more accurately reflects the two pump operation. This change does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 nor does it require a change to the Technical Specifications. Therefore, prior NRC approval was not required.

Issued: March 30, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-036 Revision 0

TEMPORARY LEAD SHIELDING - REACTOR FLANGE SEATING SURFACE INSPECTION/REPAIR

Temporary lead shielding was installed inside the Unit 3 containment upper cavity. The shielding was designed to reduce the dose rates received by personnel performing work. The temporary lead shielding was installed with the unit in Mode 6 with the reactor head on its storage stand on the 58 foot elevation.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that; there were no detrimental effects on plant operation and safety; and that no Technical Specification changes were required; and that the change supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding installed when the unit was in Mode 6 did not require any changes to the Plant Technical Specifications and did 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: May 8, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-044 Revision 0

LOW PRESSURE STEAM TURBINE LOOSE BLADES

This safety evaluation supported the continued use of loose low pressure steam turbine blades, found during a recent in-service inspection. Although the design of the blades and their mountings is such that the blade can not be loose at operating speeds, a sealing compound is used to hold the blades secure when not in motion. This sealing compound was found to be missing from some of the blades.

Safety Evaluation Summary:

This evaluation concluded that the absence of sealing compound on blade life and the use of loose blades during normal operation does not involve an unreviewed safety question as defined in 10 CFR 50.59 criteria or a change to the Technical Specifications. There are no safety concerns. There are no unreviewed safety question, and no changes are required to the Technical Specifications.

Issued: April 3, 1990

SAFETY EVALUATION: JPN-PTN-SEES-90-049 Revision 0

TEMPORARY USE OF DIGITAL MW METER FOR SAFEGUARD TESTING

This safety evaluation supports the temporary installation of digital kilowatt meters on Unit 4 for the Emergency Diesel Generators (EDG) 4A and 4B. The meters are used for EDG load testing data collection.

Safety Evaluation Summary:

The installation of the digital kilowatt meters has no adverse effects on plant safety or operations. This installation involved neither an unreviewed safety question nor a change to the Technical Specification. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: April 17, 1990

SAFETY EVALUATION: JPN-PTN-SENJ-90-051 Revision 3

PLANT CONFIGURATION DURING EMERGENCY POWER SYSTEM ENHANCEMENT

This safety evaluation demonstrates that the plant configuration required during the dual unit outage does not result in an unreviewed safety question or require changes to the Technical Specifications. This evaluation also demonstrated that the required configurations remain within the safety and licensing bases and pose no increase in risk to the health and safety of the public.

Safety Evaluation Summary:

Based on the conclusions of this safety evaluation, the plant restrictions and actions required, the plant configuration during the dual unit outage with all EDGs out of service and both reactor cores off-loaded is acceptable and does not compromise the safety and licensing bases for Turkey Point Units 3 and 4. This safety evaluation demonstrated that the plant configuration required during the dual unit outage did not result in an unreviewed safety question or require changes to the Technical Specifications. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: September 18, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-053 Revision 0

TEMPORARY INSTALLATION OF A PRESSURE INDICATOR (GAUGE) ON THE WASTE DISPOSAL SYSTEM OUTSIDE CONTAINMENT

A pressure gauge was installed below Waste Disposal System drain valve CV-4-4668D. The gauge was installed to monitor the emptying of the Reactor Coolant Drain Tank (RCDT). This installation provided a manual means of indicating when the RCDT pumps needed to be stopped due to low level in the RCDT in lieu of the normally automatic operation with LT-4-1003. This is a temporary installation until the next Unit 4 outage of adequate length when LT-4-1003, the RCDT level transmitter, can be repaired.

This change allows Operations to determine when the RCDT is empty and therefore minimize the amount of time that the RCDT pumps are run while cavitating. This change has no adverse effect on the plant's margin of safety or operability.

The RCDT only acts as a drain collection point and this change allows operations to better monitor its level and ensure no effluent backs up into the Reactor Coolant Pump (RCP) stand pipe. As such, this change places no restrictions on the operations of the plant. This system is classified as Quality Related. This change resulted in a temporary revision to operating procedure 4-OP-061.3 which details the use of the temporary gauge.

This temporary modification is downstream of containment isolation valves, CV-4-4668 A & B. The drain valve, CV-4-4668D, is normally closed and is procedurally opened for approximately 3 minutes every 100 hours to monitor pump pressure.

Safety Evaluation Summary:

The addition of a small mass to this piping system has a negligible effect on the system seismic qualification. This portion of the system has no safety or shutdown function except the closure of containment isolation valves CV-4-4668 A & B. This modification is on a branch line downstream of the containment isolation valves therefore, it will have no effect on the ability to close those valves. This change does not constitute an unreviewed safety question or a change to any Technical Specification pursuant to 10 CFR 50.59. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: May 11, 1990

SAFETY EVALUATION: JPN-PTN-SEES-90-056 Revision 0

INSTALLATION OF NEEDLE VALVES FOR FT-605, FLOW TRANSMITTER FOR RESIDUAL HEAT REMOVAL DISCHARGE TO COLD LEG

Temporary needle valves were installed between the existing instrument valve manifold and flow transmitter FT-605, flow transmitter for the residual heat removal discharge to the cold leg. The needle valves will act as pulsation dampening to minimize excessive flow process pulsations which were causing spurious low flow nuisance alarms in the Control Room.

Safety Evaluation Summary:

This evaluation shows that the installation did not constitute an unreviewed safety question, did not require a change to any Technical Specification, and did not impact plant operation or safety. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: December 19, 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-061 Revision 0

Temporary System Alteration (TSA) for Cleaning of the Turbine Lube Oil for Turkey Point Unit 4

A temporary 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 temporary filtration system was installed and may be operated during any mode of reactor operation without adversely impacting the operation of any system in Unit 3 or Unit 4.

Safety Evaluation Summary:

Based on this evaluation, this change did not constitute an unreviewed safety question or a change to the plant Technical Specification pursuant to 10 CFR 50.59.

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

Issued: May 24, 1990

SAFETY EVALUATION: JPN-PTN-SEES-90-062 Revision 0

RESTORATION OF R-18 CHANNEL, WASTE DISPOSAL SYSTEM LIQUID EFFLUENT MONITOR, TO SERVICE

A Temporary System Alteration used an alternate cable to replace the damaged cable for the instrumentation drawer of R-18, Waste Disposal System Liquid Effluent Monitor.

<u>Safety Evaluation Summary:</u>

Based on this evaluation, this change does not constitute an unreviewed safety question or a change to the plant Technical Specification pursuant to 10 CFR 50.59.

This change has no effect on plant operation and no safety concerns. Thus prior NRC approval was not required for implementation.

Issued: May 25, 1990

INSTALLATION OF ALTERNATE SFP COOLING SYSTEM COMPONENTS

The alternate Spent Fuel Pit Cooling System (SFPCS) was installed as a temporary backup for the normal SFPCS. The temporary system will remain installed until the end of the dual-unit outage. The major components of this alternate SFPCS are one (1) flat plate heat exchanger, one (1) evaporative cooling tower, and two (2) circulating pumps. The piping associated with the temporary SFPCS was installed on each unit to facilitate ready hook-up.

Safety Evaluation Summary:

The purpose of this safety evaluation is to address the acceptability of the temporary SFPCS piping and component installation and operation for potential adverse interaction with existing safety related structures, systems or components.

The alternate SFPCS serves no safety function. The normal SFPCS is not required to have active redundancy. However, the alternate system will be installed temporarily as a readily available backup method to provide SFP cooling capacity in case the normal SFPCS is rendered inoperable. The system is sized to provide 100% cooling capacity for the pre-dual-unit outage SFP heat load, and partial heat removal capacity for the SFP heat load during the dual unit outage. Based on field inspection, the location of the alternate SFPCS piping and components are such that their failure will not result in adverse interactions with existing safety related systems, structures, or components. Therefore, the piping and equipment are considered as Class III components. As stated earlier, the temporary alternate SFPCS serves no safety functions, and there are no adverse seismic interactions with safety related equipment. Therefore, this safety evaluation was classified as Not Safety Related.

Revision 0 of the safety evaluation addressed the location of the alternate SFPCS for seismic interactions and placed plant restrictions based on the need for a walkdown.

Revision 1 removes the restrictions of revision 0 based on the evaluation of the walkdown results and verification of wind loading design adequacy, and addresses the operation of the alternate SFPCS.

Revision 2 provides the basis for addressing the alternate return line connection to the temporary SFPCS. The results and conclusions of the safety evaluation remain unaffected by this revision.

INSTALLATION OF ALTERNATE SFP COOLING SYSTEM COMPONENTS

Based upon the unresolved safety question determination, the safety evaluation concludes that location and operation of the alternate SFPCS components do not result in an unresolved safety question, do not require a change to the Technical Specifications, and do not impact plant safety or safe operation. Therefore, this activity did not require prior NRC review and approval.

Issued: August 28, 1990

SAFETY EVALUATION JPN-PTN-SENJ-90-073 Revision 2

OPERATION OF THE POST ACCIDENT CONTAINMENT VENTILATION SYSTEM

In February of 1990, the Post Accident Containment Ventilation (PACV) System was found isolated for greater than seven days, which is a reportable event. During review of the event, the NRC questioned the capability to operate this system due to potentially high dose rates in the vicinity of the PACV system during post accident conditions. FPL committed to evaluate this situation and reported the results in a supplemental LER.

Safety Evaluation Summary:

Evaluation of this situation shows that while doses are high, the PACV system could be safely placed in service prior to containment hydrogen concentration levels exceeding three percent by volume. The evaluation, however recommends that corrective action in the form of procedural changes be taken to reduce the potential exposure to operators from this system post-accident. These procedure changes have been determined not to create an unreviewed safety question or require a change to the Technical Specifications, and do not impact plant operation or safety. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: October 31, 1990

SAFETY EVALUATION JPN-PTN-SEMJ-90-074 Revision 1

FISHER CONTROLS ANOMALY NOTICE (FAN) 89-1

Fisher Controls Anomaly Notice (FAN) 89-1 was issued on June 15, 1989, informing FPL of a potential problem with "micro-flow or micro-flute" valve trim material. When the valve disk and seat ring are fabricated of 316 stainless steel, evidence of galling between the disk and seat have been found. This galling can potentially affect the ability of the valve to be stroked by the actuator.

The purpose of this safety evaluation is to determine if any valves have been supplied to Turkey Point Units 3 and 4 that meet the criteria of FAN 89-1.

Safety Evaluation Summary:

This evaluation shows that two safety related valves installed at Turkey Point Units 3 and 4 met the criteria of FAN 89-1. However, further evaluation (JPN-PTN-SEMJ-90-076) determined that for those two valves, the condition described in FAN 89-1 does not constitute a substantial safety hazard. This condition does not affect the Turkey Point Units 3 and 4 Technical Specifications, and does not impact plant operation or safety.

Issued: December 21, 1990

SAFETY EVALUATION JPN-PTN-SEMJ-90-075 Revision 1

FISHER CONTROLS ANOMALY NOTICE (FAN) 89-2

Fisher Controls Anomaly Notice (FAN) 89-2 was issued on June 15, 1989, informing FPL of a potential problem with wear on top mounted handwheels when incorrectly used on several Fisher actuator types. Adjusting the handwheel while under full spring load without instrument air to position the valve could cause premature wear and early failure of the head screw threads.

On March 1, 1982, Fisher changed the design to incorporate a standard Acme thread. This change extends the useful life of the handwheel to at least 300 full stroke cycles under full spring load.

The purpose of this safety evaluation is to determine if any valves have been supplied to Turkey Point Units 3 and 4 that meet the criteria of FAN 89-2.

<u>Safety Evaluation Summary:</u>

This evaluation shows that no safety related valves installed at Turkey Point Units 3 and 4 met the criteria of FAN 89-2. Thus this condition does not affect the Turkey Point Units 3 and 4 Technical Specifications, and does not impact plant operation or safety.

Issued: December 21, 1990

SAFETY EVALUATION JPN-PTN-SEMJ-90-076 Revision 1

FISHER CONTROLS ANOMALY NOTICE (FAN) 89-1 DETERMINED TO BE APPLICABLE TO REACTOR COOLANT DRAIN TANK NITROGEN PRESSURE REGULATING VALVES PCV-3-1014 AND PCV-4-1014

Fisher Controls Anomaly Notice (FAN) 89-1 was issued on June 15, 1989, informing FPL of a potential problem with "micro-flow or micro-flute" valve trim material. This trim, when the valve disk and seat ring are fabricated of 316 stainless steel, has shown evidence of galling the disk and seat which can potentially affect the ability of the valve to be stroked by the actuator.

The purpose of this safety evaluation was to determine if Reactor Coolant Drain Tank nitrogen pressure regulating valves PCV-3-1014 and PCV-4-1014, meeting the criteria of FAN 89-1, constituted a Substantial Safety Hazard.

Safety Evaluation Summary:

This evaluation determined that the condition described in FAN 89-1 does not constitute a substantial safety hazard. This condition does not affect the Turkey Point Units 3 and 4 Technical Specifications, and does not impact plant operation or safety. The evaluation did recommend changing the material of the valve disk guide tip from 316 stainless steel to a hardened material.

Issued: December 21, 1990

SAFETY EVALUATION JPN-PTN-SENS-90-084 Revision 0

INSTALLATION OF TEMPERATURE PROBES IN THE SPENT FUEL POOL TO SUPPORT TP-642

Turkey Point performed a special test by securing all cooling to the Unit 4 spent fuel pool (SFP) to determine the heat up rate. The SFP has installed permanent temperature indication; but data on potential SFP temperature stratification when the SFP cooling system is secured was also desired. Therefore additional temperature probes were installed in the SFP, consisting of thermocouples wired together and weighted, and then suspended over the pool curb into the SFP. The test equipment was installed and operated per the requirements of TP-642, "Suspension of SFP Cooling."

Safety Evaluation Summary:

This evaluation shows that the installation of temporary temperature probes and the acquisition of temperature data in the SFP did not constitute an unreviewed safety question, did not require a change to any Technical Specification, and did not impact plant operation or safety. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: August 24, 1990

SAFETY EVALUATION: JPN-PTN-SENS-90-088 Revision 2

INSTALLATION OF FIRE HOSE BONNET ADAPTERS TO CCW VALVES 772 AND 776

The plant installed bonnet adapters to the Component Cooling Water (CCW) isolation valves to the SFP heat exchangers. These bonnet adapters (stab in's) allow for emergency (back-up) connection of fire hoses to provide for heat removal from Unit 3 and 4 spent fuel pools. This capability is desired during the dual unit outage as a back-up in the event of a loss of CCW, since full core off loads and corresponding high heatup rates of the SFP's are applicable. The high heatup rates limit the amount of SFP cooling system down time before SFP boiling would occur.

The SFP cooling system does not include emergency power as a design requirement and the SFP boiling analysis demonstrates that off-site doses will remain well within 10 CFR 100 limits, therefore, the SFP cooling function and ICW/CCW support function are considered as quality related functions. The back-up provisions installed per a Temporary System Alteration (TSA) are not required to meet the SFP system design but are desired as conservative measures to prevent SFP boiling.

Revision 1 made editorial corrections and incorporated the latest Unit 3 fuel inventory safety evaluation. The results and conclusions of this safety evaluation remain unaffected by this revision.

Revision 2 made editorial corrections and deleted the requirement to have the reactor cavity flooded during installation or removal of stab-ins. The results and conclusions of this safety evaluation remain unaffected by this revision.

Safety Evaluation Summary:

This evaluation showed that the installation of the fire hose adapters during Mode 6 and the corresponding removal of valve internals from CCW valves 772 and 776 did not constitute an unreviewed safety question, did not require a change to any Technical Specification, and did not impact plant operation or safety. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

This evaluation also showed that the removal of the fire hose adapters during Mode 6 and the corresponding restoration of valve internals in CCW valves 772 and 776 do not constitute an unreviewed safety question, do not require a change to any Technical Specification, and do not impact plant operation or safety. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: October 31, 1990

SAFETY EVALUATION JPN-PTN-SECS-90-089 Revision 0

TEMPORARY LEAD SHIELDING - REACTOR CAVITY SHADOW SHIELDING

Health Physics requested the installation of temporary lead shielding inside the Unit 3 containment upper cavity. The shielding is designed to reduce the dose rates received by personnel performing work in the reactor cavity, from radiation generated by the CRDM coils and the reactor head components. The temporary lead shielding was installed when the reactor head was on the reactor vessel and the Reactor Coolant System (RCS) de-pressurized. This shielding shall be removed prior to RCS pressurization and prior to entering Mode 4 from Mode 5.

Safety Evaluation Summary:

The installation of the temporary lead shielding has been evaluated to ensure that there will be no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification. This installation does support ALARA for personnel radiation exposure.

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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-090 Revision 0

TEMPORARY LEAD SHIELDING - PRESSURIZER SPRAY SYSTEM

This safety evaluation covers the temporary lead shielding installed on a portion of the Pressurizer Spray System, specifically at the top of the pressurizer and adjacent to PCV-3-455A and PCV-3-455B. The shielding is designed to reduce the dose rates received by personnel performing work in the area. This shielding shall be removed prior to entering Mode 4 from Mode 5.

Safety Evaluation Summary:

The installation of the temporary lead shielding has been evaluated to ensure that there will be no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification. This installation does support ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding installed 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-091 Revision 0

TEMPORARY LEAD SHIELDING - STEAM GENERATOR MANHOLES (SECONDARY SIDE)

Temporary lead shielding was attached to the Unit 3 Steam Generator handholes inside containment at the 30 foot, 6 inch level to protect maintenance personnel from excessive radiation exposure generated when the Steam Generator handhole covers are removed. The lead shielding will be in place only with Unit 3 in Mode 5, Mode 6, or defueled with the secondary side of the steam generator(s) drained.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there will be no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding installed when the unit is in either Mode 5, Mode 6, or when the unit is defueled 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-092 Revision 0

TEMPORARY LEAD SHIELDING - STEAM GENERATOR PRIMARY MANWAY

A temporary Steam Generator (SG) Primary Manway Radiation Attenuating Door (RAD) System was installed inside Unit 4 containment at approximately elevation 25 feet to protect maintenance personnel from excessive radiation exposure on the SG platform and under the manways in preparation for, and during, Eddy Current Testing (ECT). The RAD System shall be in place only while the unit is defueled and the primary side is drained. It shall be removed and the steam generator manway reinstalled prior to refilling the primary side and entering Mode 6.

Safety Evaluation Summary:

The installation of the RAD System was evaluated to ensure that there would be no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the installation of the RAD System when the unit was defueled did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-093 Revision 0

TEMPORARY LEAD SHIELDING - REGENERATIVE HEAT EXCHANGER AND LCV-3-460

Temporary lead shielding was installed inside the Unit 3 containment at elevation 14 feet between the regenerative heat exchanger and LCV-3-460, and for the piping upstream and downstream of the valve. This shielding was designed to reduce the dose rates generated by the regenerative heat exchanger and LCV-3-460.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there was no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification.

This safety evaluation concluded 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-094 Revision 0

STRUCTURAL ASSESSMENT OF THE UNITS 3 AND 4 INTAKE STRUCTURE

The intake structure at Turkey Point Nuclear Plant, Units 3 and 4, is a reinforced concrete structure consisting of eight bays. Each unit has three bays which house intake cooling water pumps and one bay containing screen wash facilities. A circulating water pump is also located in each of the bays. During outages in the 1987-1989 time period, several cracks, spalls, and other areas of degradation were noted and repaired. This safety evaluation evaluates the current condition and the continued operation of the Units 3 and 4 intake structure.

Safety Evaluation Summary:

The current conditions of the Unit 3 and 4 intake structures are not expected to seriously degrade in the near future and based on calculations, inspections, and the current assessments of each individual bay, the structure satisfies all design basis requirements. Upon completion of the mitigative actions during the dual unit outage and the implementation of the composite beam fix in each of the bays during subsequent refueling outages, it is estimated that the service of the intake structure will be extended beyond the presently projected plant life with only minimum maintenance.

This evaluation determined that this condition did not have a detrimental effect on plant operation and safety. This condition does not affect any Technical Specification. The safety evaluation also concluded that this condition does not require any changes to the Plant Technical Specifications, and does not affect plant safety or operation pursuant to 10 CFR 50.59.

Issued: September 6, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-096 Revision 0

TEMPORARY LEAD SHIELDING - REACTOR CAVITY SHADOW SHIELDING

Temporary lead shielding was installed inside the Unit 4 containment upper cavity. This shielding was designed to reduce the dose rates received by personnel, performing work in the reactor cavity, from radiation generated by the Control Rod Drive Motor coils and the reactor head components. This shielding shall be removed prior to Reactor Coolant System (RCS) pressurization and prior to entering Mode 4 from Mode 5.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety and no affect on any Technical Specification.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-097 Revision 0

TEMPORARY LEAD SHIELDING - PRESSURIZER SPRAY SYSTEM

Temporary lead shielding was installed on a portion of the Pressurizer Spray System. This shielding was designed to reduce the dose rates received by personnel performing work in the area.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety. This installation did not affect any Technical Specification.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-098 Revision 0

TEMPORARY LEAD SHIELDING - STEAM GENERATOR HANDHOLES (SECONDARY SIDE)

Temporary lead shielding was attached to the Unit 4 Steam Generator (SG) handholes inside containment at elevation 30 feet 6 inches to protect maintenance personnel from excessive radiation exposure generated when the SG handhole covers are removed. The lead shielding shall be in place only with Unit 4 in Mode 5, Mode 6, or defueled with the secondary side of the steam generator(s) drained.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety. This installation did not affect any Technical Specification.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-099 Revision 0

TEMPORARY LEAD SHIELDING - STEAM GENERATOR PRIMARY MANWAY

A temporary Steam Generator (SG) Primary Manway Radiation Attenuating Door (RAD) System was installed inside Unit 4 containment at approximately elevation 25 feet to protect maintenance personnel from excessive radiation exposure on the SG platform and under the manways in preparation for, and during, Eddy Current Testing (ECT). The RAD System shall be in place only while the unit is defueled and the primary side is drained. It shall be removed and the steam generator manway reinstalled prior to refilling the primary side and entering Mode 6.

Safety Evaluation Summary:

The installation of the RAD System was evaluated to ensure that there would be no detrimental effect on plant operation and safety. This installation does not affect any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the installation of the RAD System did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-100 Revision 0

TEMPORARY LEAD SHIELDING - REGENERATIVE HEAT EXCHANGER AND LCV-4-460

Temporary lead shielding was installed inside the Unit 4 containment at elevation 14 feet between the regenerative heat exchanger and LCV-4-460 and for the piping upstream and downstream of the valve. The shielding is required to reduce the dose rates generated by the heat exchanger and LCV-4-460 and piping while working on LCV-4-460.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety and no affect on any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-101 Revision 0

TEMPORARY LEAD SHIELDING - CVCS CHARGING AND LETDOWN LINES (200/300 SERIES VALVES)

Temporary lead shielding was installed on a portion of the Chemical and Volume Control System (CVCS). The portion of the system being shielded included the piping adjacent to the charging valves CV-4-310A, CV-4-310B, and CV-4-311, and the piping adjacent to the letdown valves CV-4-200A, CV-4-200B, and CV-4-200C (including the valve bodies). The valves are located outside the bio-shield wall in a heavily travelled area at the 14 foot level of the containment building. The shielding was required to reduce the dose rates generated in the area of the valves.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there were no detrimental effect on plant operation and safety and no affect on any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: November 15, 1990

SAFETY EVALUATION: JPN-PTN-SEES-90-102 Revision 1

TEMPORARY SYSTEM ALTERATION TO BYPASS THE OVERTRAVEL LIMIT SWITCH ON THE SPENT FUEL CASK BRIDGE CRANE

The overtravel limit switch has failed and restricts use of the Spent Fuel Cask Bridge Crane. Until a replacement switch is obtained, Turkey Point installed a jumper to bypass the limit switch and allow continued operation of the crane. The limit switch in question provides an automatic interlock to prevent the crane from challenging the mechanical stops at the north end of the crane rail. Although during normal operation, crane travel is limited such that the limit switch is not often actuated. If the administrative controls initiated fail to stop the crane, the mechanical stops will limit crane overtravel and subsequent damage. In addition, a deadman switch will de-energize the control circuit, thus stopping the crane if a failure occurred in the control circuit.

<u>Safety Evaluation Summary:</u>

The overtravel limit switch was not intended to restrict the movement of loads over the spent fuel storage areas as discussed in the Technical Specifications. The switch should only affect crane travel when the crane is at the end of the rail. This evaluation demonstrates that this change did not involve an unreviewed safety question or a change to the Technical Specifications. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: September 20, 1990

SAFETY EVALUATION: JPN-PTN-SEES-90-103 Revision 0

INTAKE COOLING WATER (ICW) SYSTEM INTERNAL INSPECTION GUIDELINES

This safety evaluation covers the inspection, cleaning, and patching work in the Unit 3 and Unit 4 ICW systems piping during the 1991 dual unit outage. ICW system operations during the dual unit outage are governed by Safety Evaluation JPN-PTN-SENJ-90-051, applicable Technical Specifications and associated system operating procedures. The work was implemented with the reactors defueled and all spent fuel stored in the spent fuel pool. A corrosion engineering firm, performed an internal crawl-through inspection, including appropriate cleaning and repair of the ICW system during the 1991 dual unit outage.

Safety Evaluation Summary:

The inspection, cleaning, and repair work performed in the Unit 3 and 4 ICW systems piping did not adversely impact plant safety and operation, did not constitute an unreviewed safety question or require a change to the governing Technical Specifications. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: December 13, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-107 Revision 0

TEMPORARY LEAD SHIELDING - REACTOR VESSEL HEAD LIFTING RIG

Temporary lead shielding was installed inside the Unit 4 containment to the upper platform of the Reactor Vessel Head Lifting Rig. The temporary lead shielding was provided to protect maintenance personnel from excessive radiation exposure generated by the CRDM coil and components. The lead shielding shall be in place while the unit is in Mode 5, 6, or defueled and shall be removed prior to entering Mode 4.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety and no affect on any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: October 19, 1990

SAFETY EVALUATION: JPN-PTN-SENJ-90-108 Revision 0

SAFETY EVALUATION FOR TURKEY POINT UNITS 3 & 4 P&ID RECONSTITUTION PROJECT

This project was the development of a new set of Piping and Instrumentation Diagrams (P&IDs) that are computer aided design produced, unit specific, expanded to include specialty drawings, enhanced and reconfigured with additional sheets to provide clarity as well as room for future updates.

Safety Evaluation Summary:

This project was classified as nuclear safety related because many of the P&IDs are for safety related systems. The safety evaluation concluded that no unreviewed safety question was involved. There is no impact on plant safety or system operational requirements. Also, the project did not require a change to any plant technical specification. Therefore, the project was implemented without prior NRC approval under the provisions of 10 CFR 50.59.

Issued: December 19, 1990

SAFETY EVALUATION JPN-PTN-SEES-90-112 Revision 1

STATIONARY SCREEN FOR THE INTAKE STRUCTURE

PC/M 89-207 was issued to provide a stationary screen for use in the intake structure bays. The screen is utilized to provide a debris barrier during periods when the traveling screens are out of service. The PC/M engineering evaluation uses, as a premise, that the affected circulating water (CW) pump will be taken out of service during the period when the stationary screen is installed. In order to facilitate repairs to the traveling water screen mechanisms, it has been requested that engineering evaluate the possibility of operating the affected CW pump with the stationary screen installed.

Revision 1 to this safety evaluation allows for the Intake Cooling Water (ICW) pump in the affected bay to remain in service during operation of the CW pump. During periods of time when the stationary screen must be manually cleaned, both the ICW pump and CW pump will be declared out of service.

<u>Safety Evaluation Summary:</u>

This safety evaluation determined that the operation of the CW pump, in the affected bay, with the stationary screen installed, and under the limitations specified, did not require any changes to the Plant Technical Specifications and did not affect plant safety or operation. Therefore, prior NRC approval was not required pursuant to 10 CFR 50.59.

Issued: October 18, 1990

SAFETY EVALUATION JPN-PTN-SENJ-90-113 Revision 0

SAFETY EVALUATION FOR INTAKE COOLING WATER (ICW) PUMP FOUNDATIONS

This evaluation was performed to determine if nonconforming anchorages for the 3A Intake Cooling Water (ICW) pump involved a substantial safety hazard. This evaluation was necessary to determine 10 CFR 21 reportability.

Non-conformance Report NCR-86-336 identified the 3A Intake Cooling Water (ICW) pump anchorages as not conforming to the applicable design documentation. Subsequent investigation revealed that additional ICW pumps had nonconforming anchorages. The condition was analyzed in Safety Evaluation JPES-C-86-8 and was determined not to involve an operability concern. The pump anchorages were subsequently repaired.

Safety Evaluation Summary:

This evaluation concluded that the deviation related to the ICW base plate attachments did not represent a Substantial Safety Hazard.

The pumps would have remained functional for both normal operating and accident conditions even with the degraded supports. Although several of the pump anchorages may have failed under seismic or hurricane conditions, required cooling could have been maintained by cross-connecting CCW between units. Previous evaluations have shown that the units are capable of reaching and maintaining Mode 3 even if all ICW is lost. Therefore this condition does not meet the 10 CFR 21 definition of a Substantial Safety Hazard.

Issued: October 18, 1990

SAFETY EVALUATION JPN-PTN-SENJ-90-114 Revision 1

OPERABILITY ASSESSMENT OF COMPONENT COOLING WATER SYSTEM FOR SPLIT HEADER CONFIGURATIONS

As a consequence of a recent review of the Component Cooling Water (CCW) pump inservice testing procedures 3-OSP-030.1 and 4-OSP-030.1, the acceptability of splitting the two CCW headers during pump inservice flow testing and other operations requiring split CCW headers was addressed. This evaluation focused on single electrical failures, because the power supplies for the three CCW pumps and certain Engineered Safety Feature (ESF) systems, such as the three Emergency Containment Coolers (ECCs), cannot be configured to serve the needs of a fully redundant and automatic two-train fluid system. Licensee Event Report (LER) 3-90-021-00 was submitted to the NRC on November 13, 1990, to report the splitting of the two CCW headers without entering the appropriate Limiting Condition for Operation (LCO) on October 18, 1990.

Safety Evaluation Summary:

The operation of the ICW and CCW systems in the split header configurations for inservice testing should have been accompanied by entry into the appropriate LCO Action statement for equipment out-of-service. However, procedural controls would have ensured restoration of systems to their normal "open" configuration in the event of an accident. With such operator actions, the system configurations would have provided heat removal within their respective design bases.

The CCW system operating in a split header configuration for movement of the cask in the cask washdown area resulted in the plant operating outside its design basis. Although the plant was operating outside its design basis, this did not result in a significant safety concern for the following reasons:

1) based on sensitivity studies performed, it is not expected that the minimum containment design pressure would be exceeded;

2) there is a high probability of restoring AC power to the nuclear units after one (1) hour;

3) Systematic Design Investigation Program risk based methodologies were applied and the resultant probability for the dominant failure scenario is very low and not a significant contributor to the overall risk of a core melt.

Issued: October 29, 1990

SAFETY EVALUATION: JPN-PTN-SEMJ-90-115 Revision 0

PRESSURIZER SURGE LINE THERMAL STRATIFICATION (BULLETIN 88-11)

On December 20, 1988, the NRC issued Bulletin 88-11 regarding pressurizer surge line thermal stratification. FPL is participating in a program which has been implemented to address the issue of thermal stratification.

Safety Evaluation Summary:

This evaluation concludes that the Justification for Continued Operation (JCO) forwarded to the NRC in FPL letter number L-89-194 is still valid. Visual inspections of the pressurizer surge lines revealed superficial damage to the insulation consistent with the expected thermal movements considering stratification effects. Also a variable spring support in Unit 3 was shown to be bottomed out, however, the support showed no signs of damage. Several cracks in the saddle-to-pipe welds were found in Unit 4. These welds serve only to hold the saddle in place and do not provide load carrying capacity to the support. The variable spring assembly did not display any signs of damage.

This safety evaluation concluded that the as-found condition of the pressurizer surge lines is acceptable, with no adverse impact on plant safety, security, or operation. This as-found condition did not create an unreviewed safety question or require a change to the Technical Specifications.

Issued: November 13, 1990

SAFETY EVALUATION: JPN-PTN-SECS-90-123 Revision 0

TEMPORARY REMOVAL OF STEAM GENERATOR THRUST BEAM AND FLOOR GRADING/STEEL FOR RIGGING REACTOR COOLANT PUMP (RCP) MOTORS

This evaluation reviewed the temporary removal of steam generator thrust beams and floor grading/steel for rigging RCP motors. The effects on existing systems, structures, and components due to the temporary removal of these structural items were evaluated with respect to plant operational modes. No permanent change in plant configuration was involved. The structural items removed will be reinstalled to the same configuration and to the same design requirements as the original installation.

Safety Evaluation Summary:

This safety evaluation concluded that removal of these structural items is acceptable, with no adverse impact on plant safety, security, or operation. This change did not create an unreviewed safety question or require a change to the Technical Specifications as demonstrated in the Safety Evaluation. Therefore, prior NRC approval pursuant to 10 CFR 50.59 was not required for implementation.

Issued: November 28. 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-125 Revision 0

USE OF AUTOMATIC START CAPABILITY FOR THE BACKUP DIESEL DRIVEN AIR COMPRESSORS AND OPERATION OF THE ELECTRIC DRIVEN AIR COMPRESSOR

This evaluation supports the use of the existing automatic start capability for the diesel driven air compressors and the operation of one electric driven air compressor. Presently, these diesel driven air compressors are operated manually, providing the primary source of instrument air required by the plant.

There are no specific plant operating restrictions associated with the use of the automatic start feature for the diesel driven air compressors, nor will this affect plant operations or safety. The operation of the electric driven air compressor has no impact on the emergency diesel generator loading or on the Appendix R commitments.

Safety Evaluation Summary:

This safety evaluation concluded that the activity addressed in the safety evaluation did not have an adverse effect on plant safety, security, or operation, did not constitute an unreviewed safety question, and did not require changes to the TS. Therefore, prior NRC approval pursuant to 10 CFR 50.59 was not required for implementation.

Issued: December 21. 1990

SAFETY EVALUATION: JPN-PTN-SECJ-90-128 Revision 0

SAFETY EVALUATION FOR REACTOR HEAD LIFT RIG ADAPTOR ASSEMBLY FOR LOAD CELL INSTALLATION

In order to provide continued compliance with NRC NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Florida Power and Light (FP&L) modified the Reactor Head Lift Rig with the installation of a direct reading tension load cell. This load cell is available for both reactor head lift operations and reactor internals lift operations. In implementing this modification, FP&L is following the vendor's recommendation.

<u>Safety Evaluation Summary:</u>

The addition of the internals load cell and the load cell linkage adapter will have no adverse effects on the operation of the lifting rig. The addition of the load cell linkage adapter and load cell assembly does not alter the function of the lifting rig and, in fact, enhances its adaptability and reliability.

This safety evaluation concluded that the activity addressed in the safety evaluation did not have an adverse effect on plant safety or operation, did not constitute an unreviewed safety question, and did not require changes to the Technical Specifications. Therefore, prior NRC approval pursuant to 10 CFR 50.59 was not required for implementation.

Issued: December 4. 1990

SAFETY EVALUATION JPN-PTN-SENS-90-130 Revision 0

ENGINEERING ASSESSMENT FOR UNIT 3 ACCUMULATOR OPERABILITY

On November 16, 1990, another licensee reported a defect in their Safety Injection Accumulators to the NRC under 10 CFR 21. The reportable defect deals with a material deviation involving the accumulator couplings (used as nozzles), that would render the couplings susceptible to inter-granular stress corrosion cracking (IGSCC). The licensee found evidence of through-wall cracks in a coupling, and additional indications of IGSCC on other couplings were identified using non-destructive testing techniques. Turkey Point Units 3 and 4 accumulators were also manufactured by the same manufacturer and supplied by the same vendor. This issue is being evaluated for applicability to Turkey Point.

The accumulators are passive components of the Emergency Core Cooling System and are required for the initial re-flood of the core following a maximum hypothetical accident. Therefore, the accumulators are classified as Safety Related.

Safety Evaluation Summary:

This safety evaluation concluded that continued operation of Unit 3 until its normally scheduled shutdown was acceptable and did not have an adverse effect on plant safety, security, or operation; did not constitute an unreviewed safety question; and did not require changes to the TS. Therefore, prior NRC approval pursuant to 10 CFR 50.59 was not required for implementation.

Issued: November 30. 1990

SAFETY EVALUATION: JPN-PTN-SEMS-90-133 Revision 0

TEMPORARY LEAD SHIELDING - UNIT 3 REACTOR COOLANT LOOPS

Temporary lead shielding was installed on a portion of the Reactor Coolant System (RCS) loops (all three loops). The temporary lead shielding was required to reduce the dose rates generated in the area.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there were no detrimental effects on plant operation and safety and no affect on any Technical Specification. The installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: December 19, 1990

SAFETY EVALUATION JPN-PTN-SEMS-90-134 Revision 0

TEMPORARY LEAD SHIELDING - UNIT 4 REACTOR COOLANT SYSTEM LOOPS

Temporary lead shielding was installed on a portion of the Reactor Coolant System (RCS) loops (all three loops). The temporary lead shielding was required to reduce the dose rates generated in the area.

Safety Evaluation Summary:

The installation of the temporary lead shielding was evaluated to ensure that there would be no detrimental effect on plant operation and safety and no affect on any Technical Specification. This installation supports ALARA for personnel radiation exposure.

This safety evaluation concluded that the temporary lead shielding did not require any changes to the Plant Technical Specifications and did 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: December 19, 1990

SAFETY EVALUATION JPN-PTN-SENS-90-135 Revision 0

ULTRASONIC INSPECTION OF NUCLEAR FUEL AT TURKEY POINT UNIT 4

The ultrasonic inspection of nuclear fuel at Turkey Point Unit 4, is considered a test not described in the FSAR.

Safety Evaluation Summary:

The proposed test, Ultrasonic Inspection of Nuclear Fuel, has been evaluated for Technical Specification changes and for unresolved safety questions in accordance with 10 CFR 50.59. The safety evaluation concludes that this does not require a change to the Technical Specifications pursuant to 10 CFR 50.59 and does not adversely impact plant safety or safe operations. Therefore, prior NRC approval was not required.

Issued: December 12, 1990

SAFETY EVALUATION JPN-PTN-SEMS-90-139 Revision 0

TEMPORARY LAUNDRY FACILITY LIQUID EFFLUENT

The FSAR states that all liquid waste components, except for the reactor coolant drains and the pressurizer relief tanks, are located in the radwaste handling building and the auxiliary building. This safety evaluation evaluated the acceptability of the deviation from the FSAR by placing the temporary laundry facility outside the radwaste building and the auxiliary building in the RCA.

The temporary laundry facility dryer vent and ventilation is filtered by HEPA filters. The HEPA filtered vents are sampled weekly and analyzed by chemistry to ensure the HEPA filters are functioning properly and are within Technical Specifications limits for particulate release.

The temporary laundry facility is considered to be Not Safety Related. The temporary laundry facility and the transfer hose, which is connected to the Molybdate Holding Tank during transfer, have no safety function nor do they interact with any safety related structures, components or systems.

Safety Evaluation Summary:

Based upon the unreviewed safety question determination, the safety evaluation concludes that the temporary laundry facility liquid effluent transfers to the radwaste system did not result in an unreviewed safety question. The safety evaluation also concluded that this process change did not require a change to the Technical Specifications pursuant to 10 CFR 50.59 and did not adversely impact plant safety or safe operations. Therefore, prior NRC approval was not required.

Issued: December 17, 1990

SAFETY EVALUATION JPN-PTN-SENS-90-140 Revision 0

FUEL ASSEMBLY REPAIR/RECONSTITUTION

This safety evaluation addressed the repair/reconstitution of standard fuel assemblies and 15 X 15 optimized fuel assemblies at Florida Power and Light Company's Turkey Point Unit 4 plant This evaluation assessed the potential safety impact of the fuel repair/reconstitution to be implemented. This evaluation was completed in accordance with 10 CFR 50.59 (a) (2) criteria.

Safety Evaluation Summary:

The safety evaluation shows that the repair/reconstitution of fuel assemblies at Turkey Point Unit 4 did not have a deleterious effect on the health and safety of the public nor did it create any unreviewed safety questions pursuant with 10 CFR 50.59 (a) (2) criteria.

Issued: December 18. 1990

SAFETY EVALUATION JPN-PTN-SECS-90-141 Revision 0

PROPOSED USE OF PROTECTIVE COATINGS IN SERVICE LEVEL 1 APPLICATIONS

This safety evaluation addresses the proposed use of protective coating systems for service level 1 applications at Turkey Point Units 3 and 4. The use of the proposed systems is desired primarily due to the fact that the proposed system is much less difficult to apply than the currently used system.

<u>Safety Evaluation Summary:</u>

The safety evaluation showed that this change did not have a deleterious effect on the health and safety of the public nor did it create any unreviewed safety questions pursuant with 10 CFR 50.59 (a) (2) criteria.

This change did not require any deviations related to design or operating practices and philosophy. This change did not require alteration of the Technical Specifications and did not constitute an unreviewed safety question. The use of the proposed systems did not adversely impact plant operation or safety and did not result in any restrictions on plant operation.

Issued: January 3, 1991

SAFETY EVALUATION: JPN-PTN-SEES-90-142 Revision 0

TEMPORARY INSTALLATION AND OPERATION OF SPARE SOURCE RANGE DETECTORS

Two spare source range detectors have been installed within the active region of the core. These detectors will be connected to the existing preamplifiers only in case of failure of either primary source range detector

Safety Evaluation Summary:

The implementation of this temporary installation with the subsequent connection of the spare detector in case of single failure, does not change plant operation or safety, and does not create an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval was not required to install the spare source range detectors and is not needed prior to connecting the spare detectors to the existing preamplifiers.

Issued: December 21. 1990

SAFETY EVALUATION: JPN-PTN-SENS-90-145 Revision 0

USE OF TEMPORARY HOSE TO SUPPORT ALTERNATE LIQUID RADWASTE RELEASE POINT

A temporary hose was used to support an alternate liquid radwaste release point when no circulating water pumps are available during the dual unit outage. The valve bonnet and internals from valve 3-50-001 were removed and a bonnet adapter was installed. A manual valve was connected to the bonnet adapter, and a high pressure service hose was routed from the bonnet adapter to the Intake cooling water discharge well.

Safety Evaluation Summary:

The installation and use of this temporary hose and connection did not change plant operation or safety, and did not create an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: January 29. 1991

SAFETY EVALUATION: JPN-PTN-SEEP-91-002 Revision 1

GENERAL ELECTRIC IAV UNDERVOLTAGE RELAY SETPOINTS

Turkey Point Florida Power and Light Quality Assurance Audit QAO-PTN-90-039, Finding 2, identified a concern regarding the setpoints for the General Electric IAV undervoltage relays. The specific concern identified was that the relays may not be repeatable with respect to the specific setpoints and tolerances. In response to this concern, the calibration frequency has been increased from 18 months to 12 months. The increase in the frequency of the calibrations provides a greater confidence that the relay setpoints remain within acceptable tolerances. This safety evaluation was performed to determine the acceptability of this corrective action.

Safety Evaluation Summary:

The safety evaluation concluded that this change provided a reasonable assurance that the relay setpoints will not drift outside their allowable tolerances. This procedural change does not have an adverse effect on plant safety, security, or operation, and does not constitute an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: June 27, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-003 Revision 0

CONTINUOUS PRESSURIZER SPRAY FOR RCS COOLDOWN

This safety evaluation was performed by the nuclear steam supply vendor to evaluate the use of a pressurizer continuous spray when the RCS to pressurizer differential temperature is 200°F $< \Delta T <$ 320°F to aid in plant cooldowns to reduce the potential for exceeding the heatup/cooldown limits. In addition this evaluation addressed two transients that occurred at Turkey Point.

Safety Evaluation Summary:

The vendor calculations determined that a continuous pressurizer spray could be utilized in lieu of the fourth and fifth spray transients without imposing additional stresses during cooldown. Note that the first three (60) sixty second sprays are required prior utilizing continuous spray.

The safety evaluation concluded that this change does not have an adverse effect on plant safety, security, or operation, and does not constitute an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: April 5, 1991

SAFETY EVALUATION: JPN-PTN-SEIJ-91-008 Revision 0

PROCEDURAL CHANGES TO INCORPORATE TECHNICAL SPECIFICATION (TS) SETPOINT CHANGES PRIOR TO FINAL NRC APPROVAL OF TS CHANGES

In an effort to reduce the critical path time associated with the numerous procedural and setpoint changes produced by the Setpoints licensing amendment, a proposal has been made to update the applicable procedures and implement the setpoint changes as applicable while the units are defueled or in Modes 6 and 5 and restrict the units to those modes until the approved amendment is issued by the NRC. The setpoints are not needed until the unit enters Mode 4.

Safety Evaluation Summary:

The safety evaluation concluded that with these mode restrictions, these setpoint changes do not affect plant safety, and do not create an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval is not required for implementation. Note, the licensee amendment request in question was initiated while the units were still defueled. The actions described in this safety evaluation enabled the licensee to make sufficient setpoint change modifications to prevent this task from becoming critical path.

Issued: May 8, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-010 Revision 0

REACTOR VESSEL STUD TENSIONING/DETENSIONING

In an effort to reduce the critical path time associated with tensioning and detensioning the reactor head, an evaluation was conducted enabling a reduction in the number of passes in the tensioning/detensioning procedure. After reviews with applicable personnel and performance of stress analysis, a reduction was recommended.

Safety Evaluation Summary:

The safety evaluation concluded that this change did not affect plant operation or safety, and did not create an unreviewed safety question or require a change any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: May 3, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-011 Revision 0

REDUCTION IN THE REACTOR COOLANT SYSTEM (RCS) MINIMUM HYDROGEN CONCENTRATION FOR MODE 2 OPERATIONS

During normal power operations, a hydrogen concentration is maintained in the primary coolant to scavenge oxygen. This evaluation considered the change of the minimum concentration from 25 to 15 cc/kg prior to entering Mode 2, while maintaining the minimum concentration of 25 cc/kg for Mode 1 operations. This change will reduce the time required to go from Mode 3 to Mode 2 during start-ups.

Safety Evaluation Summary:

The safety evaluation concluded that this change did not affect plant operation or safety, and did not create an unreviewed safety question or require a change to any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: March 13, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-015 Revision 2

RAW WATER BOOSTER PUMP REPLACEMENT

This safety evaluation reviews the effects of a Turkey Point Units 1 and 2 fossil plant initiated modification that affects a system shared by both nuclear and fossil sites, including the nuclear side fire protection system. This evaluation will serve as a basis to revise applicable nuclear plant procedures.

Safety Evaluation Summary:

The safety evaluation concluded that this change did not affect plant operation or safety, and did not create an unreviewed safety question or require a change to any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: June 7, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-017 Revision 0

EVALUATION OF TESTING REQUIREMENTS FOR HEPA FILTERS LOCATED IN THE EXHAUST VENTS OF THE AUXILIARY BUILDING, SPENT FUEL BUILDINGS, RADWASTE BUILDING AND CONTAINMENT PURGE SYSTEM

This safety evaluation reviews the requirements for HEPA filter testing requirements in light of the physical impossibility of performing some of the test. The Auxiliary Building, Spent Fuel Buildings, and Radwaste Building ventilation systems, and the Containment Purge System were installed before the testing requirements were developed and no physical modifications were made to the systems to allow performance of these tests.

Safety Evaluation Summary:

The safety evaluation concluded that the current program of accelerated replacement of the HEPA filters is sufficient to insure operability of the filters located in the exhaust vents of the Auxiliary Building, Spent Fuel Buildings, Radwaste Building, and Containment Purge System. The only advantage the tests would provide is the operability assurances that would permit the filters to remain in use for a longer period of time.

The safety evaluation concluded that the current program does not have an adverse effect on plant safety, security, or operation, and does not constitute an unreviewed safety question or change any Technical Specifications.

Issued: April 5, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-024 Revision 0

COMPONENT COOLING WATER HEAT EXCHANGER TUBE SUBSTANTIAL SAFETY HAZARD EVALUATION

This safety evaluation was provided to determine if the recent Turkey Point specific issues regarding Component Cooling Water heat exchanger tube material deficiencies are a substantial safety hazard as defined in 10 CFR 21. After tube failures, material testing identified the failed tubes as being manufactured from admiralty brass instead of the aluminum brass as required.

Safety Evaluation Summary:

The safety evaluation concluded that a substantial safety hazard as defined by 10 CFR 21 does not exist for the component cooling water heat exchangers due to the use of the wrong material. This conclusion was based on the actual tube failures experienced during this event and the design of the component cooling water system along with the Technical Specifications that govern the operation of this system.

Issued: April 5, 1991

SAFETY EVALUATION: JPN-PTN-SEMS-91-025 Revision 0

HYDROGEN FROM HOLDUP TANKS PURGE TO ATMOSPHERE

To reduce the potential for a hydrogen explosion, the holdup tanks will be purged into the Auxiliary Building HVAC system prior to opening associated piping and valves. Samples indicated a 6 percent hydrogen presence and no activity in the gas in the holding tanks.

<u>Safety Evaluation Summary:</u>

The safety evaluation concluded that the proposed purge rates do not result in an unreviewed safety question. The safety review also concluded that the proposed purge rates do not change plant operation or safety, and do not create an unreviewed safety question or change any Technical Specifications. Therefore, prior NRC approval was not required for implementation.

Issued: March 22. 1991

SAFETY EVALUATION JPN-PTN-SECS-91-039 Revision 1

THE USE OF UNITS 1 & 2 TURBINE GANTRY CRANE ON UNITS 3 & 4 TURBINE OPERATING DECK

This safety evaluation addresses the use of the Turkey Point Units 1 & 2 Turbine Gantry Crane on the Units 3 & 4 turbine operating deck while the nuclear units are defueled. The crane will be returned to the fossil side of the plant prior to the refueling of either Unit 3 or Unit 4.

Safety Evaluation Summary:

The temporary use of the Units 1 & 2 Turbine Gantry Crane on the Units 3 & 4 turbine operating deck does not change the design bases, functions, or operations of any safety related equipment and does not adversely affect any other safety related structures, systems, or components. This modification was reviewed against the requirements of 10 CFR 50.59 and determined to have no adverse impact on plant safety, security, or operation. Therefore, prior NRC approval was not required for this change.

Issued: June 14, 1991

SAFETY EVALUATION JPN-PTN-SENS-91-043 Revision 1

EMERGENCY DIESEL GENERATOR BACKUP TO C-BUS DURING EMERGENCY POWER SYSTEM ENHANCEMENT

Revision 3 of the Dual Unit Outage safety evaluation imposes a restriction, that two blackstart diesels be maintained functional to ensure on-site AC power availability. To support a specific sequence of activities during the dual unit outage, it is advantageous to use an emergency diesel generator as the on-site backup AC power source in lieu of the blackstart diesels. This safety evaluation evaluated the acceptability of this alternative and verified that the availability of reliable power was not compromised.

Safety Evaluation Summary:

The temporary use of one of the EDGs as the on-site backup AC power source did not change the design bases, functions, or operations of any safety related equipment and did not adversely affect any other safety related structures, systems, or components. This modification was reviewed against the requirements of 10 CFR 50.59 and determined to have no adverse impact on plant safety, security, or operation. Therefore, prior NRC approval was not required for this change.

Issued: June 6, 1991

SAFETY EVALUATION JPN-PTN-SEMS-91-045 Revision 0

TEMPORARY ALTERATION FOR THE HIGH VELOCITY OIL FLUSH OF THE TURBINE LUBE OIL SYSTEM

A temporary filtration system was installed to reduce particulates in the Unit 4 Turbine Lube Oil System and the Unit 4 Generator Seal Oil Systems. This temporary system was installed and operated in accordance with a temporary procedure. The high velocity oil flush was installed and operated while Unit 4 is out of service for the Dual Unit Outage. The high velocity oil flush had no adverse impact on any safety related system for Unit 3 or Unit 4.

Safety Evaluation Summary:

The evaluation showed that this temporary system did not adversely affect equipment addressed in the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specifications was not reduced since the basis of any Technical Specification was not affected.

Thus, this evaluation demonstrates that the temporary system alteration did not involve an unreviewed safety question or require a change to the Technical Specification pursuant to 10 CFR 50.59, and prior NRC approval for this activity was not required.

Issued: June 14, 1991

SAFETY EVALUATION JPN-PTN-SENS-91-047 Revision 0

KEYWAY GATE REMOVAL DURING DUAL UNIT OUTAGE

The Unit 3 spent fuel pool keyway gate seal experienced an increase in leakage that necessitated its removal from the keyway for repair. The repair required that the gate be lifted entirely out of the spent fuel pool. Although previous lifts of this gate had been made, the current safety evaluation in effect during this dual unit outage prohibits heavy load lifts over the spent fuel pool during this outage. This evaluation assessed the proposed gate lift and provided a basis for a temporary waiver of compliance.

Safety Evaluation Summary:

This evaluation provided a reasonable degree of assurance that the lift would be safe and that the probability of a load drop was extremely remote. Based on this evaluation, this change did not constitute an unreviewed safety question or a change to the plant Technical Specification pursuant to 10 CFR 50.59.

Thus, this evaluation demonstrated that this lift did not involve an unreviewed safety question or change to the Technical Specification pursuant to 10 CFR 50.59. However, due to the possibility that the lift may bring the gate over spent fuel, prior NRC approval for this activity was required and obtained.

Issued: June 6, 1991

SECTION 3

RELOAD SAFETY EVALUATIONS

RELOAD SAFETY EVALUATIONS

SAFETY EVALUATION: NF-90-330

TURKEY POINT UNITS 3 AND 4 CYCLE 12 RELOAD - REVISION OF SHUTDOWN MARGIN

Post trip RCS cooldowns of as much as 27 degrees Fahrenheit below the estimated no-load T_{avg} have been experienced at Turkey Point. If these cooldowns had occurred near the end of cycle, shutdown margin requirements could have been violated. As a result, the fuel vendor was requested to review the shutdown margin formulation to determine if excessive conservatism exists, which could be reduced.

As a result of the vendor review, the Turkey Point Units 3 and 4 cycle 12 reload shutdown margins have been revised. This revision reduced the conservatisms from the axial reactivity redistribution allowance and control rod uncertainty allowance for Turkey Point Unit 3, cycle 12 (BOL and EOL) and Unit 4, cycle 12 (EOL only) Reload Safety Evaluation.

Safety Evaluation Summary

This safety evaluation concluded that the implementation of the revision to the Reload Safety Evaluation for Turkey Point Units 3 and 4 , cycle 12 reloads did not involve any changes which introduce an unreviewed safety question. Therefore, implementation of this revision to the shutdown margin calculations for the reload was permissive without prior NRC approval.

Note: No core reloads were performed during the reporting period, July 1, 1990 to June 30, 1991.

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 3 and 4 from July 1, 1990 to June 30, 1991.

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

December 13, 1990	PORV 455C and 456 were cycled per 3-OP-041.4

Unit 4

July 21, 1990	PORV 455C and 456 were cycled per 4-OP-041.4
July 21, 1990	PORV 455C and 456 was cycled per OP 209.1
July 23, 1990	PORV 456 was cycled per 4-OSP-041.4
July 30, 1990	PORV 456 was cycled per 4-OSP-041.4
July 30, 1990	PORV 455C and 456 were cycled per 4-OSP-041.4

July 31, 1990	PORV 455C was cycled per 4-OSP-041.4
July 31, 1990	PORV 455C was cycled per 4-OSP-041.4
November 26, 1990	PORV 455C cycled as a result of actuation of the Overpressure Mitigation System
November 26, 1990	PORV 455C and 456 were cycled per 4-OP-041.4

SECTION 5

STEAM GENERATOR TUBE INSPECTIONS FOR UNIT 4

Eddy Current Summary of Results					
Plant: Tu~key Point 4					
Examination Dates: 4/16/91 Through 5/13/91					
Steam Generator Number	Total Tubes Inspected	Total Ind. ≥ 20% to 39%	Total Ind. ≥ 40% to 100%	Total Tubes Plugged as Preventive Maintenance	Total Tubes Plugged
4E210A	3198	13	None	1	1
4E210B	3207	10	None	2	2
4E210C	3205	8	None	None	None

Location of Indications

Steam Generator	AVB Bars	Drilled Support 1 through 6		Top of Tube Sheet to 1 Drilled Support	
		Cold Leg	Hot Leg	Cold Leg	Hot Leg
4E210A	None	4	8	1	None
4E210B	1	2	4	1	2
4E210C	None	4	3	None	1

Certification of Record

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

FLORIDA POWER and LIGHT COMPANY
Organization

Date: 07/08/91 By: KR Craig
Manager, Heat Exchanger and Met Lab

Steam Generator Tubes Plugged

Plant: Turkey Point 4

Steam Generator 4E210A			Steam Generator 4E210B			Steam Generator 4E210C		
Row	Column	Remarks	Row	Column	Remarks	Row	Column	Remarks
2	5	Replace hot & cold	8	81	Restricted Tube	-	NONE	-
			31	14	Replace hot leg			

Steam Generator Eddy Current Examination Results

Plant: Turkey Point 4 Steam Generator: 4E210A
Examination Dates: 4/16/91 through 5/13/91

Row	Column	% Tube Wall Penetration	Origin	Location
28	14	32	OD	01H + 42.4
-	-	34	OD	02C + 2.6
33	19	23	OD	05H + 42.8
26	24	29	OD	03H - 0.5
29	25	33	OD	04H + 5.8
-	-	37	OD	BAC + 28.3
27	40	23	OD	05C + 25.8
33	42	21	OD	05H + 41.5
26	62	26	OD	05C + 17.7
30	76	26	OD	05H + 46.0
24	77	20	OD	02H + 47.4
14	82	26	OD	04C + 9.2
12	83	23	OD	01H + 40.5

Steam Generator Eddy Current Examination Results				
Plant: Turkey Point 4 Examination Dates: 4/16/91			Steam Generator: 4E210B through 5/13/91	
Row	Column	% Tube Wall Penetration	Origin	Location
13	10	29	OD	06H - 0.5
8	18	24	OD	TSC + 3.8
3	24	27	OD	03C + 25.5
40	35	34	OD	BAH + 12.2
45	48	21	OD	AV4 + 0.0
44	51	35	OD	04H + 20.4
37	69	36	OD	TSH + 22.0
-	-	35	OD	02H + 19.1
14	82	26	OD	02H + 15.9
24	85	38	OD	05C + 27.8

Steam Generator Eddy Current Examination Results

Plant: Turkey Point 4 Steam Generator: 4E210C
Examination Dates: 4/16/91 th·ough 5/13/91

Row	Column	% Tube Wall Penetration	Origin	Location
28	28	28	OD	05H + 44.3
-	-	30	OD	06H - 0.7
11	32	33	OD	TSH + 2.2
3	52	36	OD	01C + 29.8
24	56	24	OD	02C + 36.6
24	62	23	OD	02C + 35.4
23	77	30	OD	02C + 50.8
13	90	20	OD	01H + 17.6

RELIEF REQUEST NO. VR-2

SYSTEM:

Steam Generator - Aux. Feedwater Supply (5610-T-E-4062-3)

COMPONENTS:

CV-*-2816	CV-*-2831
CV-*-2817	CV-*-2832
CV-*-2818	CV-*-2833

CATEGORY:

B

FUNCTION:

These valves open to provide flowpaths from the auxiliary feedwater pumps to the respective steam generators.

SECTION XI REQUIREMENT:

If, for power-operated valves, an increase in stroke time of 50% or more for valves with full-stroke times less than or equal to 10 seconds is observed, the test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed (IWV-3417(a))

BASIS FOR RELIEF:

These valves open with nitrogen or instrument air pressure from a positioner signal to control auxiliary feedwater flow to the steam generators. In the event of a loss of air or electric power to the valve control system, they will fail to the closed position. When a valve is closed it can be opened by pressure from the associated AFW pumps, thus it affords no isolation to prevent over-feeding the steam generators and therefore these valves have no specific safety function in the closed position.

BASIS FOR RELIEF (cont.):

The initial opening signal for these valves comes from a limit switch in the associated AFW pump steam supply motor-operated valve(s). After a valve opens, its position is determined by the automatic flow control system or with a manual controller located in the Control Room. Due to the automatic functioning of the valves and the absence of position indication devices, there is no practical mechanism for accurately measuring valve stroke time.

These valves are subjected to periodic testing that verifies proper operation of the valves upon initiation of AFW system initiation and proper response and positioning of each valve to a respective control system output air signal. These tests provide a high level of confidence that the valves will perform their safety function as intended.

Stroke times of these valves are determined by adjusting the manual controller in the Control Room while a local observer at the valve measures the movement time of the valve stem. The stroke time measurements taken during testing of these valves are expected to be less than 10 seconds. Due to the relative speed of the valves and consideration of the method of measurement of these stroke times, the test data is subject to considerable variation due to conditions unrelated to the material condition of the valve (eg. test conditions, operator reaction time, communication lag).

The proposed alternate testing along with the additional testing performed outside the scope of the IST Program will provide adequate assurance that these valves will perform, as required, with a high degree of reliability.

ALTERNATE TESTING:

The stroke time for these valves will be determined but the evaluation of the stroke times will not account for successive increases of measured stroke time per IWV-3417(a) with the change in test frequency as required. In lieu of this, an assigned maximum limiting value of stroke time will be established consistent with the operational requirements for the valves and of the AFW system and the stroke time history of the valves when they are known to be operating satisfactorily. Upon exceeding that limit, a subject valve will be declared inoperable and corrective action taken in accordance with IWV-3417(b).

RELIEF REQUEST NO. VR-6

SYSTEM:

Chemical & Volume Control (CVCS) (5610-T-E-4505-1&3)

COMPONENTS:

HCV-*-0121

CATEGORY:

A

FUNCTION:

These valves open to provide flowpaths from the charging pumps to the reactor coolant system during emergency boration. They close for containment isolation.

SECTION XI REQUIREMENT:

If, for power-operated valves, an increase in stroke time of 50% or more for valves with full-stroke times less than or equal to 10 seconds is observed, the test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed (IWV-3417(a))

BASIS FOR RELIEF:

These valves are subjected to periodic testing that verifies proper operation of the valves and their response and positioning with respect to control system output air signal. These tests providea high degree of confidence that the valves will perform their safety function as intended.

RELIEF REQUEST NO. VR-6 (cont.)

BASIS FOR RELIEF (cont.):

These valves are positioned from the Control Room by manually adjusting a DC electric current signal input to a current-to-air pressure converter that transmits a predetermined air pressure to the valve positioner that adjusts valve position and the associated charging and reactor coolant pump seal injection flows. In the event of a loss of air or electric power to the valve control system, they will fail in the open position. Since there is no position indication or specific actuating signal for these valves to effect valve operation, measuring an accurate stroke time per IWV-3413 is not practical - however, local observation of valve movement and operation is an effective way of ascertaining the material condition of the valves.

Stroke times of these valves are determined by adjusting the manual controller in the Control Room while a local observer at the valve measures the movement time of the valve stem. Due to the expected relatively high speed of the valves and consideration of the method of measurement of these stroke times, the test data is subject to considerable variation due to conditions unrelated to the material condition of the valve (eg. test conditions, operator reaction time, communication lag).

ALTERNATE TESTING:

The stroke time for these valves will be determined but the evaluation for these valves will not account for successive increases of measured stroke time per IWV-3417(a) with the change in test frequency as required. In lieu of this, an assigned maximum limiting value of stroke time will be established consistent with the operational requirements for the valves and of the CVCS system and the stroke time history of the valves when they are known to be operating satisfactorily. Upon exceeding that limit, a subject valve will be declared inoperable and corrective action taken in accordance with IWV-3417(b).

RELIEF REQUEST NO. VR-9

SYSTEM:

Safety Injection (5610-T-E-4510-1)

COMPONENTS:

3-0890 A&B
4-0890 A&B

CATEGORY:

A/C

FUNCTION:

These valves open to provide flowpaths from the containment spray pumps to the containment spray headers in containment. They are required to close for containment isolation.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Full-stroke exercising these valves to the open position would require operating each containment spray pump at nominal accident flowrate. Since no recirculation flowpath exists downstream of these valves, the only flowpath available for such a test would result in injecting radioactive-contaminated borated water into the containment spray headers and thence into the containment building via the spray nozzles. Dousing personnel and equipment in this manner is obviously undesirable.

BASIS FOR RELIEF (cont.)

Each of these valves has been disassembled and inspected in the past and they have not displayed any indication of degradation that would impede their capability to perform their safety function to open. Past inspections were conducted as follows with no indication of a valve's inoperability with respect to its capability to full open:

3-0890A	4-7-90
3-0890B	3-24-90
4-0890A	3-18-89 and 3-8-91
4-0890B	3-18-89 and 3-8-91

Partial stroking of the valves can be achieved by pressurizing the upstream piping with air or nitrogen via the air test connection. This, however, results in the possibility of creating an airborne contamination condition in the auxiliary building or containment and, furthermore, would result in minimal valve disc movement. Hence, such testing would be of little value in determining valve operability.

These valves remain closed at all times except when the core spray system operates during an incident for containment cooling and de-pressurization.

Since these are simple-acting check valves with no provision for determining disc position, the only practical means of verifying closure involves performing a leaktest. Performance of such a test at each cold shutdown requires isolation of the containment spray headers and a valve alignment to allow for pressurizing the downstream piping with air or nitrogen and venting the upstream piping. If this activity were required to be performed during cold shutdown periods it would constitute an unreasonable burden on the plant staff that is not justified by any benefit gained from verifying the valves have remained closed when there is no reason to believe otherwise.

ALTERNATE TESTING:

During each reactor refueling outage at least one of these valves in the associated unit will be disassembled, inspected, and manually exercised on a sequential and rotating schedule. If, in the course of this inspection a valve is found to be inoperable with respect to its function to fully open, then the other valve in the same unit will be inspected during the same outage.

Following valve disassembly, the subject valve(s) will be partial-stroked in the open direction with air or nitrogen pressure followed by a seat leakage test to verify proper functioning.

Each of these valves will be verified to be closed at least once every two (2) years in conjunction with Appendix J leaktesting activities.

RELIEF REQUEST NO. VR-11

SYSTEM:

Safety Injection (SIS) (5610-T-E-4510-2)

COMPONENTS:

3-0874 A&B
4-0874 A&B

CATEGORY:

A/C

FUNCTION:

These valves open to provide flowpaths for borated water injection from the SIS pumps to "A" and "B" RCS hot legs. Additionally, they close to provide isolation of the SI system from the RCS high pressure.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Because no recirculation path exists downstream of these valves, exercising these valves requires operating a safety injection pump and injecting into the reactor coolant system. At power operation this is not possible because the SIS pumps cannot develop sufficient discharge pressure to overcome reactor coolant system pressure. During normal cold shutdown conditions, injection via the SIS pumps is precluded by operational restrictions related to low-temperature over-pressurization protection concerns and the related Technical Specifications.

BASIS FOR RELIEF (cont.)

Since these are simple-acting check valves with no provision for determining disc position, the only practical means of verifying closure involves performing a leaktest. Performance of such a test at each cold shutdown requires containment entry and typically partial draining of the safety injection piping. Performing these leak tests on a quarterly-based schedule would constitute an unreasonable burden on the plant staff. The Technical Specifications, Section 4.4.6.2.2, establishes a more appropriate frequency for leak testing based on their pressure isolation function. The Technical Specification requirements are adequate to confirm valve operability in the closed position. The requirements of the Technical Specifications are as follows:

* At least once every 18 months;
* Prior to entering Mode 2 whenever the plant has been in cold shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months;
* Prior to returning a valve to service following maintenance, repair, or replacement work on the valve; and
* Following valve actuation due to automatic or manual action or flow through the valve:
    1. Within 24 hours by verifying valve closure, and
    2. Prior to entering Mode 2 by verifying valve leakrate

ALTERNATE TESTING:

At least once during each reactor refueling outage, each of these valves will be full-stroked exercised to the open position.

Valve closure testing will conform to the requirements of Turkey Point Technical Specification, Section 4.4.6.2.2.

RELIEF REQUEST NO. VR-12

SYSTEM:

Safety Injection (SIS) (5610-T-E-4510-2)

COMPONENTS:

3-0875 A-C
4-0875 A-C

CATEGORY:

A/C

FUNCTION:

These valves open to provide flowpaths for borated water injection from the SIS pumps, the RHR pumps, and the SIS accumulators to each of the RCS cold legs. Additionally, they close to provide isolation of the SI system from the RCS high pressure.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Full stroke exercising of these valves to the open position, based on the maximum accident flowrate resulting from SIS accumulator injection to a de-pressurized RCS loop is not practical due to limitations associated with the effects of such a test on system components.

BASIS FOR EXTENDED INSPECTION INTERVAL (cont.):

of any one of these valves would result in an estimated 6.5 man-rem, constituting a considerable increase in personnel exposure.

In 1988, all three valves in Unit 4 and in 1990 one valve in Unit 3 were disassembled and inspected, and after approximately 15 years of plant operation for each valve, they were found to be fully operable and in excellent condition. Based on this history, an inspection interval of 10 years is sufficient to ensure continued operability of these valves.

These six valves have been reviewed against the EPRI installation guidelines and it was predicted that under service conditions, the disc will oscillate at low levels with low fatigue and wear indices.

A review of industry experience was performed using the INPO NPRDS failure history database with the following results:

* A total of 90 failures of safety injection check valves was reported including both combined injection path valves and typically identical accumulator discharge valves.

* A total of 46 failures (51%) were a result of seat leakage.

* A total of 35 failures (39%) were a result of leakage of the body-to-bonnet joint (ie. gasket failure) with no effect on the safety function of the valves.

* All other failures (9) were of a design or wear nature where conditions were identified as a result of mandated inspections or inspections initiated by the plant staff for other concerns. Six of these are related to cracking of the retainer block studs on Anchor Darling valves.

* Of the total, failures of Anchor Darling valves similar to those installed at Turkey Point, represented 33 failures.

* None of the failures identified would have resulted in a valve being unable to fully open or perform its safety function to open.

BASIS FOR RELIEF (cont.):

It has been demonstrated, by testing, that these valves can be fully opened by blowdown from a partially pressurized (100 psig.) accumulator to the associated RCS loop. During such a test, unobtrusive testing techniques could be applied to verify full stroking of these valves. This method, however, presents safety problems when performed during periods when fuel is in the reactor. These concerns include the potential for dislodging fuel assemblies and gas binding of the residual heat removal pumps. Either of these concerns precludes routine performance of accumulator blowdown tests without off-loading the core. Imposition of a requirement for core off-load and accumulator dump testing during each reactor refueling outage would constitute an unjustified burden on the plant staff and would result in a significant and unwarranted impact on plant availability.

The maximum flowrate achievable by means other than accumulator discharge is approximately 4000 gpm developed by two RHR pumps injecting into a de-pressurized reactor coolant system. This flowrate results in a flow velocity of approximately 20 feet per second (fps) equal to approximately 40% of the peak flowrate expected during accumulator injection. The valve manufacturer's data indicates that 20 fps. is "approximately" that flowrate required to open these valves. Due to the lack of sufficient margin whereby full stroke of these valves can be assured at this flowrate, it is questionable as to the capability of consistently full-stroking these valves with this limited flowrate such that non-intrusive testing could be employed effectively and reliably.

Due to the system configuration, the total flow from the two RHR pumps 4000 gpm can be directed through *-875A alone; however, in the case of valves *-875B and *-875C, the flow is split between the two valves, theoretically 2000 gpm through each valve. This is clearly inadequate to fully open the "B" and "C" valves.

Because no downstream recirculation path exists partial-flow testing of these valves requires injecting into the RCS. At power operation this is not possible because neither the RHR or the SIS pumps can develop sufficient discharge pressure to overcome reactor coolant system pressure. During normal cold shutdown conditions, however, injection via the RHR pumps can be accomplished.

BASIS FOR RELIEF (cont.):

Since these are simple-acting check valves with no provision for determining disc position, the only practical means of verifying closure involves performing a leaktest. Performing leaktests on these valves requires containment entry and opening normally closed test line isolation valves along with partial draining of safety injection piping. Conducting this testing on a quarterly-based schedule would constitute an unreasonable burden on the plant staff. The Technical Specifications, Section 4.4.6.2.2, establishes a more appropriate frequency for leak testing based on their pressure isolation function. The Technical Specification requirements are adequate to confirm valve operability in the closed position. The requirements of the Technical Specifications are as follows:

* At least once every 18 months;

* Prior to entering Mode 2 whenever the plant has been in cold shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months;

* Prior to returning a valve to service following maintenance, repair, or replacement work on the valve; and

* Following valve actuation due to automatic or manual action or flow through the valve:

    1. Within 24 hours by verifying valve closure, and

    2. Prior to entering Mode 2 by verifying valve leakrate

BASIS FOR EXTENDED INSPECTION INTERVAL:

Disassembly and inspection of any of these valves will require defueling of the reactor and drain down below mid-nozzle level or mid-loop shutdown cooling operations. If done during each reactor refueling outage, this would result in an unacceptable and unnecessary burden on the plant staff. In addition, based on past experience, an inspection

RELIEF REQUEST NO. VR-12 (cont.)

BASIS FOR EXTENDED INSPECTION INTERVAL (cont.):

From the foregoing, it is clear that full-flow testing would not have identified any of the reported conditions, and, by far, the most sensitive means of identifying valve deterioration is leaktesting. Furthermore, the data also suggests that more frequent disassembly may not be prudent in that it may tend to increase the already high failure rate of body-to-bonnet gaskets.

During each reactor refueling, each of these valves is partial-stroked to the open position and then subjected to a leaktest. If a valve were deteriorated to the extent that operability related to its capability to pass the required flow were impaired, it is highly unlikely that a successful leakrate could be performed.

ALTERNATE TESTING:

Each of these valves will be partial stroke tested to the open position during cold shutdown in accordance with Paragraph 4.6 and following reassembly for those valves subjected to disassembly and inspection.

At least once during each 10-year inspection interval all six valves will be full stroke exercised. If exercising and verification of full stroke is not practical during the interval, each valve will be disassembled, inspected, and manually exercised per Reference 2.8, Position 2.

Valve closure testing will conform to the requirements of Technical Specification, Section 4.4.6.2.2

RELIEF REQUEST NO. VR-13

SYSTEM:

Safety Injection (SIS) (5610-T-E-4510-2)

COMPONENTS:

3-0875 D-F
4-0875 D-F

CATEGORY:

A/C

FUNCTION:

These valves open to provide flowpaths for borated water injection from the SIS accumulators to each of the RCS cold legs. They close to isolate the SIS accumulators from reactor coolant system pressure and to prevent diversion of flow from the safety injection paths into a partially full accumulator.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Full stroke exercising of these valves to the open position, based on the maximum accident flowrate resulting from SIS accumulator injection to a de-pressurized RCS loop is not practical due to limitations associated with the effects of such a test on system components.

Since these are simple-acting check valves with no provision for determining disc position, the only practical means of verifying closure involves performance of a leaktest. Performing leaktests on these valves requires containment entry and opening normally closed test line isolation valves. Conducting this testing on a quarterly-based schedule would constitute an unreasonable burden on the plant staff.

BASIS FOR RELIEF (cont.)

It has been demonstrated by testing that these valves can be fully opened by blowdown from a partially pressurized (100 psig.) accumulator to the associated RCS loop. During such a test, unobtrusive testing techniques could be applied to verify full stroking of these valves. This, however, presents safety problems when performed during periods when fuel is in the reactor. These concerns include the potential for dislodging fuel assemblies and gas binding of the residual heat removal pumps. Either of these concerns precludes routine performance of accumulator blowdown tests without off-loading the core. Imposition of a requirement for core off-load and accumulator dump testing during each reactor refueling outage is an unjustified burden on the plant staff and would result in a significant and unwarranted impact on plant availability.

BASIS FOR EXTENDED INSPECTION INTERVAL:

Disassembly and inspection of these valves will require defueling of the reactor and drain down below mid-nozzle level or mid-loop shutdown cooling operations. If done during each reactor refueling outage, this would result in an unacceptable and unnecessary burden on the plant staff.

These valves have been disassembled several times in the recent past without noting any degradation that would indicate that a valve would not fully open. The recent history of disassembly is as follows:

3-0875D	1990 and 1991
4-0875D	1989
4-0875E	1989
4-0875F	1988

These valves are essentially identical to those addressed in Relief Request VR-12 and the industry failure history database for these valves is included in that presented in Relief Request VR-12.

BASIS FOR EXTENDED INSPECTION INTERVAL (cont.):

Again from the available failure history, it is clear that full-flow testing would not have identified any of the reported conditions, and, by far, the most sensitive means of identifying valve deterioration is leaktesting. Furthermore, the data also suggests that more frequent disassembly may not be prudent in that it may tend to increase the already high rate of failure of body-to-bonnet gaskets.

During each reactor refueling, each of these valves is partial-stroked to the open position and then subjected to a leaktest. If a valve were deteriorated to the extent that operability related to its capability to pass the required flow were impaired, it is highly unlikely that a successful leakrate could be performed.

These valves have essentially no usage and, thus, detailed review against the EPRI installation guidelines has not been performed. However, due to the low usage factor, little or no deterioration is expected due to service conditions.

ALTERNATE TESTING:

Each of these valves will be partial-stroke tested to the open position during each reactor refueling in accordance with Paragraph 4.6 and following re-assembly for those valves subjected to disassembly and inspection. Any valve disassembled will also be subjected to a leaktest.

At least once during each 10-year inspection interval all six valves full stroke exercised. If exercising and verification of full stroke exercising is not practical during the interval, each valve will be disassembled, inspected, and manually exercised per Reference 2.8, Position 2.

During each reactor refueling, each of these valves is subjected to a leaktest to verify closure.

RELIEF REQUEST NO. VR-14

SYSTEM:

Safety Injection (SIS) (5610-T-E-4510-2)

COMPONENTS:

3-0876 B&C
4-0876 B&C

CATEGORY:

A/C

FUNCTION:

These valves open to provide flowpaths for borated water injection from the RHR pumps to "B" and "C" RCS cold legs. Additionally, they close to provide isolation of the RHR system from the reactor coolant system.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Exercising these valves requires operating an RHR pump and injecting into the reactor coolant system since no recirculation path exists. At power operation this is not possible due to system design pressure and interlocks that prevent operation of the RHR system in cooldown alignment when RCS pressure exceeds 515 psig. During normal cold shutdown conditions, injection via the RHR pumps is practical.

During cold shutdown conditions these valves can be full-stroke exercised. Since they have no position indicators and are installed such that the only lineup available causes them to form a parallel path, full accident flow through each valve cannot be confirmed as required by Reference 2.8, Position 1, and thus full stroke verification is not practical.

Note: Identical valves 3-0876A and 4-0876A are full stroked exercised during cold shutdown.

RELIEF REQUEST NO. VR-14 (Cont.)

BASIS FOR RELIEF (cont.):

Since these are simple-acting check valves with no provision for determining disc position, the only practical means of verifying closure involves performing a leaktest. Performance of such a test requires shutdown of the residual heat removal system and significant re-alignment of system valves. Conducting these tests at each cold shutdown would constitute an unreasonable burden on the plant staff. The Technical Specifications, Section 4.4.6.2.2, establishes a more appropriate frequency for leak testing based on their pressure isolation function. The Technical Specification requirements are adequate to confirm valve operability in the closed position. The requirements of the Technical Specifications are as follows:

* At least once every 18 months;

* Prior to entering Mode 2 whenever the plant has been in cold shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months;

* Prior to returning a valve to service following maintenance, repair, or replacement work on the valve; and

* Following valve actuation due to automatic or manual action or flow through the valve:

    1. Within 24 hours by verifying valve closure, and

    2. Prior to entering Mode 2 by verifying valve leakrate

BASIS FOR EXTENDED INSPECTION INTERVAL:

Disassembly and inspection of these valves will require defueling of the reactor and drain down below mid-nozzle level or mid-loop shutdown cooling operations. If done during each reactor refueling outage, this would result in an unacceptable and unnecessary burden on the plant staff.

In 1988, in Unit 4 both of these valves along with valves 4-876 D&E (identical valves) were disassembled and inspected and in 1990, in Unit 3 valves 3-876 B&D were likewise disassembled and inspected. After approximately 15 years of plant operation, all valves that were inspected were found to be fully operable and in excellent condition. Based on this history, an inspection interval of 10 years is sufficient to ensure continued operability of these valves.

These valves are similar in design and function to those addressed in Relief Request VR-13 and the industry failure history database for these valves is included in that presented in Relief Request VR-13.

Again from the available failure history, it is clear that full-flow testing would not have identified any of the reported conditions, and, by far, the most sensitive means of identifying valve deterioration is leaktesting. Furthermore, the data also suggests that more frequent disassembly may not be prudent in that it may tend to increase the already high rate of failure of body-to-bonnet gaskets.

During reactor refueling, each of these valves is partial-stroked (full-stroked without flow measurement) to the open position and then subjected to a leaktest. If a valve were severely deteriorated to the extent that operability related to its capability to pass the required flow were impaired, it is highly unlikely that a successful leakrate could be performed.

These four valves have been reviewed against the EPRI installation guidelines and it was predicted that under service conditions, the disc will oscillate with some moderate tapping. The fatigue and wear indices are calculated to be very low and low, respectively.

ALTERNATE TESTING:

Each of these valves will be partial stroke tested to the open position during cold shutdown in accordance with the provisions of Paragraph 4.6. In the event that an appropriate non-obtrusive method of verifying valve operation should become available, full stroke exercising of these valves will then be performed during cold shutdown, if practical.

At least once during each 10-year inspection interval these valves will be disassembled, inspected, and manually exercised per Reference 2.8, Position 2. Following re-assembly, partial flow and leak tests will be performed on any valve subjected to disassembly.

Valve closure testing will conform to the requirements of Technical Specification, Section 4.4.6.2.2

RELIEF REQUEST NO. VR-22

WITHDRAWN

RELIEF REQUEST NO. VR-29

SYSTEM:

Steam Generator - Aux Feedwater Supply (5610-T-E-4062-3)

COMPONENTS:

AFWU-3-0017
AFWU-4-0016

CATEGORY:

C

FUNCTION:

These valves open to provide pathways for cooling water from the auxiliary feedwater pump bearing coolers to the condensate storage tanks.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

Full-stroke exercising of these valves would require simultaneous operation of all three auxiliary feedwater pumps. Operation in such a mode during a test is not practical or desirable. In addition, there is no instrumentation available to verify flow in the line.

ALTERNATE TESTING:

During quarterly testing of the auxiliary feedwater pumps these valves will be part-stroke exercised using the flow from one pump.

During each reactor refueling outage the respective valve will be disassembled and inspected to verify freedom of motion of the disc assembly. Following re-assembly, the affected valve(s) will be partial-flow exercised.

RELIEF REQUEST NO. VR-30

SYSTEM:

Chemical and Volume Control (CVCS) (5610-T-E-4505-5)

COMPONENTS:

3-0351
4-0351
3-0397 A&B
4-0397 C&D

CATEGORY:

C

FUNCTION:

These valves open to provide pathways for emergency boration from the boric acid pumps to the charging pump suctions. They close to prevent recirculation through an idle pump.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

3-0397 A&B and 4-0397 C&D

During plant operation, due to concerns about over-borating the RCS, the boric acid pumps are tested via a recirculation flowpath that is not provided with any flow indication. Thus, since flowrate through these valves cannot be measured, in accordance with the provisions of Reference 2.8, it is considered to be a partial-stroke test. At cold shutdown conditions the pumps can be lined up to pump to the charging pumps and thus through an instrumented line, however, testing these valves in this manner would require the introduction of highly concentrated boric acid solution from the boric acid tanks to the suction of the charging pumps and, thence, to the RCS. The additional boric acid introduced into the RCS would cause considerable operational difficulty during the ensuing startup.

BASIS FOR RELIEF (cont.):

*-0351

Testing these valves requires the introduction of highly concentrated boric acid solution from the boric acid tanks to the suction of the charging pumps. This, in turn, would result in the addition of excess boron to the RCS which adversely affects plant power level and operational parameters with the potential for an undesirable plant transient and a plant trip or shutdown. During cold shutdown, the additional boric acid introduced into the RCS would cause considerable operational difficulty during the ensuing startup.

ALTERNATE TESTING:

During quarterly testing of the boric acid transfer pumps valves 3-0397 A&B and 4-0397 C&D will be part-stroke exercised using the recirculation flowpath to the boric acid tanks.

During each reactor refueling outage each of the valves in the associated unit will be full-stroke exercised.

RELIEF REQUEST NO. VR-31

SYSTEMS:

Main Steam

COMPONENTS:

3-10-004 thru 3-10-006
4-10-004 thru 4-10-006

CATEGORIES:

C

FUNCTION:

In the event of an upstream steam line break these valves close to prevent blowdown of more than one steam generator. closed position.

SECTION XI REQUIREMENT:

Check valves shall be exercised at least once every 3 months, except as provided by IWV-3522. (IWV-3521)

BASIS FOR RELIEF:

These are large stop check valves in the main steam lines leading to the main turbine generator. There is no practical way of verifying closure of these valves by way of a back seat or reverse flow test. Exercising the valve manually using the hand wheel does not confirm that the valve would close during a steam break incident but gives some assurance that the disc moves freely within the valve body. During plant operation at power closure of these valve would result in a severe transient on the plant with the chance of a plant trip.

ALTERNATE TESTING

Each of these valves will be exercised to the closed position using the handwheel during cold shutdown in accordance with the provisions of Paragraph 4.6.

RELIEF REQUEST NO. VR-31 (cont.)

ALTERNATE TESTING (cont.):

At least once during each 10-year inspection interval these valves will be disassembled, inspected, and manually exercised per Reference 2.8, Position 2. Following reassembly, the valves will be exercised to the open position during the subsequent startup.

In the event that an appropriate non-obtrusive method of verifying valve operation (closure) should become available, verification of valve closure will then be performed on a practical schedule depending on the complexity and degree of difficulty of the test method.

RELIEF REQUEST NO. VR-32

SYSTEM:

Emergency Diesel Generator (5614-M-736, Sh 2&3)

COMPONENTS:

SV-4-3434 A&B

CATEGORY:

B

FUNCTION:

These valves open to provide flowpaths for diesel fuel oil from the fuel oil transfer pumps to the respective day tanks.

SECTION XI REQUIREMENT:

If, for power-operated valves, an increase in stroke time of 50% or more for valves with full-stroke times less than or equal to 10 seconds is observed, the test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed (IWV-3417(a))

BASIS FOR RELIEF:

These valves are totally enclosed solenoid valves having no local or remote position indication nor any other means of determining valve position, thus, measuring an accurate stroke time is impractical. They are opened in response to a signal from a day tank level switch that also starts the associated pump. The only option available for taking such measurements would be to replace these valves and control circuits with those of a design that would provide an appropriate means of allowing accurate stroke time determination. The installed valves and control system are well suited to the intended service and backfitting merely for providing accurate stroke time information is considered to be unwarranted at this time.

Note that the stroke times of these valves is not critical from the aspect of accident mitigation. A qualitative assessment of valve operation by local observation will be adequate for ensuring the proper operation of the valves.

BASIS FOR RELIEF (cont.):

An estimate of valve stroke time can be obtained by inserting a "dummy" start signal from the associated day tank level control system and noting the satisfactory operation of the diesel fuel oil transfer pump and subjective flow assessment through the solenoid valves. The stroke time measurements taken during testing of these valves are expected to be on the order of 2-4 seconds. Due to the relative speed of the valves and consideration of the method of measurement of these times, the test data is subject to considerable variation due to conditions unrelated to the material condition of the valve (eg. test conditions, operator reaction time, communication lag).

ALTERNATE TESTING:

During quarterly diesel generator testing each of these valves will be exercised and verified to operate satisfactorily. The stroke time for these valves will be determined but the evaluation of the stroke times will not account for successive increases of measured stroke time per IWV-3417(a) with the change in test frequency as required. In lieu of this, an assigned maximum limiting value of stroke time will be established consistent with the operational requirements for the valve and of the diesel fuel oil system and with the stroke time history of the valves when they are known to be operating acceptably. Upon exceeding that limit, a subject valve will be declared inoperable and corrective action taken in accordance with IWV-3417(b).