FPL

APR 1 6 2001

L-2001-89
10 CFR 50.59

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

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

Florida Power and Light Company's summary report on Changes, Tests and Experiments Made Without Prior Commission Approval, for the period from April 9, 1999, through October 23, 2000, is attached. It also contains a summary of the Power Operated Relief Valve actuations, and the results of the Units 3 and 4 steam generator tube inspections, which occurred during that time. This report also includes the reload safety evaluation summaries for Unit 3 Cycle 18, dated August 30, 2000, and Unit 4 Cycle 19, dated December 31, 2000.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

DRL

Attachment

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

IE47

FPL

**Inter-Office Correspondence**

PTN-ENG-01-0097

To: S. M. Franzone

Date: April 13, 1999

From: D. J. Tomaszewski

Department: PTN Engineering

Subject: **TURKEY POINT UNITS 3&4**
**1999/2000 ANNUAL 10 CFR 50.59 SUMMARY REPORT**

Attached is the 1999/2000 Annual 10 CFR 50.59 Summary Report for Plant Management review. It includes a summary of each 50.59 safety evaluation prepared as part of a PC/M, Stand Alone Safety Evaluation, or Reload Evaluation between the period of April 9, 1999 and October 23, 2000. It also includes a summary of the PORV actuations that have occurred at Turkey Point over that period of time, and the results of any steam generator tube inspections.

All Engineering and Licensing comments received to date have been incorporated. Please forward any Plant Management comments to Engineering as soon as possible so we can meet the required NRC submittal date of 4/23/01 for this report. If you have any questions, please contact Mitch Guth at x 6698.

**Action Summary**

Submit the attached report for Plant Management review and provide comments to Engineering as soon as possible.

D. J. Tomaszewski for
PTN Engineering Manager

DJT/WAS/RJT/MCG

Attachment: 1999/2000 Annual 10 CFR 50.59 Summary Report

cc: H. S. Bowles w/o Attachment

**TURKEY POINT UNITS 3 & 4**

# EVALUATION

# OF

# CHANGES, TESTS AND EXPERIMENTS

# MADE AS ALLOWED BY 10 CFR 50.59

# FOR THE PERIOD COVERING

# APRIL 9, 1999 THROUGH OCTOBER 23, 2000

PTN-ENG-SENS-01-0032
REVISION 0

**SAFETY RELATED**

## REVIEW AND APPROVAL RECORD

PLANT Turkey Point Plant UNIT 3 & 4

TITLE Evaluation of Changes, Tests and Experiments Made as Allowed by 10 CFR 50.59 for the Period Covering April 9, 1999 Through October 23, 2000

LEAD DISCIPLINE Design Basis Group

ENGINEERING ORGANIZATION FPL - Nuclear Engineering

**REVIEW/APPROVAL:**

| GROUP | INTERFACE TYPE | | | PREPARED | VERIFIED | APPROVED | FPL APPROVED* |
|---|---|---|---|---|---|---|---|
| | INPUT | REVIEW | N/A | | | | |
| MECH | | | N/A | N/A | N/A | N/A | N/A |
| ELECT | | | N/A | N/A | N/A | N/A | N/A |
| I&C | | | N/A | N/A | N/A | N/A | N/A |
| CIVIL | | | N/A | N/A | N/A | N/A | N/A |
| DESIGN BASIS** | YES | | | [signature] 4/13/01 | Robert [signature] 4/13/01 | W Shelly | N/A |
| RRAG | | | N/A | N/A | N/A | N/A | N/A |
| SYS ENGR | | | N/A | N/A | N/A | N/A | N/A |
| COMP ENGR | | | N/A | N/A | N/A | N/A | N/A |

* For contractor Evals as determined by Projects ** Review interface as a Min on all 10CFR50.59 Evals and PLAs

**FPL PROJECTS APPROVAL:** William A. Shelly **DATE:** 4/13/01

**OTHER INTERFACES**

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## 1.0 DESCRIPTION AND PURPOSE

The purpose of this document is to provide a vehicle for the review and approval of Engineering's summary report of changes, tests, and experiments made pursuant to 10 CFR 50.59, for the period covering April 9, 1999 through October 23, 2000. This summary report is required to be submitted to the NRC in accordance with 10 CFR 50.59(d)(2).

Since the information requested by the NRC is associated with safety related plant documentation and analyses, this evaluation is classified as safety related.

## 2.0 ANALYSIS

The attached document provides a cogent summary of the Engineering Packages and stand-alone safety evaluations that were approved by the Plant Nuclear Safety Committee (PNSC) during the period covering April 9, 1999 through October 23, 2000. These documents comprise the set of plant changes, tests and experiments that were conducted during the reporting period without prior Commission approval. The established reporting period is consistent with that defined for the Revision 17 update of the Turkey Point FSAR and complies with the requirements delineated in 10 CFR 50.59(d)(2). A list of power operated relief valve (PORV) actuations that occurred during the 18-month period is also included in the attached document. This information is included as part of FPL's commitment to comply with the requirements of Item II.K.3.3 of NUREG 0737. A summary of any steam generator tube inspections conducted during the reporting period is also provided.

## 3.0 10 CFR 50.59 APPLICABILITY

This evaluation provides Turkey Point's submittal to the NRC pursuant to 10 CFR 50.59(d)(2). The information provided in this evaluation does not change the plant configuration, basis for design, or method of plant operation. It is not associated with any test or experiment, and in no way affects the Fire Protection Program, Offsite Dose Calculation Manual, or environmental plan. Thus, 10 CFR 50.59 is not applicable to this document.

## 4.0 VERIFICATION STATEMENT

This evaluation provides information for submittal to the NRC in compliance with 10 CFR 50.59(d)(2) and NUREG 0737. All 10 CFR 50.59 evaluations that were PNSC approved during the period covering April 9, 1999 through October 23, 2000 have been included in the attachment. The regulatory requirements were properly identified and the results of this evaluation meet those requirements.

The rationale provided in assigning the Safety Classification was verified against the requirements of ENG-QI 2.6, "Safety Classifications" (Reference 1). It was correctly concluded that this evaluation be classified as safety related.

## 5.0 REFERENCES

1. ENG-QI 2.6, "Safety Classifications," Revision 1

# 1999/2000 ANNUAL

# 10 CFR 50.59 SUMMARY REPORT

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 & 4

**TURKEY POINT PLANT UNITS 3 AND 4**

**DOCKET NUMBERS 50-250 AND 50-251**

**CHANGES, TESTS AND EXPERIMENTS**

**MADE AS ALLOWED BY 10 CFR 50.59**

**FOR THE PERIOD COVERING**

**APRIL 9, 1999 THROUGH OCTOBER 23, 2000**

## INTRODUCTION

This report is submitted in accordance with 10 CFR 50.59, 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 for the same period as required by 50.71(e) for the Turkey Point FSAR update. This report is intended to meet this requirement for the period covering April 9, 1999, through October 23, 2000.

This report is divided into five (5) sections. The first section summarizes those changes made to the facility as described in the SAR that were performed by a Plant Change/Modification (PC/M). The second section summarizes those changes made to the facility or procedures as described in the SAR that were performed by a Safety Evaluation. This includes those changes not performed by a PC/M, and any tests and experiments not described in the SAR that were performed during this reporting period. The third section provides a summary of the Unit 3 and Unit 4 fuel reload evaluations. The fourth section provides a list of power operated relief valve (PORV) actuations. This section is included as part of FPL's commitment to comply with the requirements of Item II.K.3.3 of NUREG 0737. The fifth and last section of this report provides a summary of the findings of any steam generator tube inspections. Both Units 3 and 4 had a steam generator tube inspection during this reporting period.

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# SECTION 1

# PLANT CHANGE / MODIFICATIONS

## PLANT CHANGE/MODIFICATION 97-022

| | | |
|---|---|---|
| UNIT | : | 4 |
| TURN OVER DATE | : | 04/15/99 |

## SAFETY INJECTION PIPE VENTING MODIFICATION

**Summary:**

This Engineering Package was developed to eliminate a personnel safety hazard associated with venting the Unit 4 high head safety injection (HHSI) piping. The venting procedure historically required that a ¾-inch blind flange be removed at valve 4-940V and a venting flange with a hose attachment be installed in its place to vent the system. The exchange had to be performed on a ladder since valve 4-940V is located approximately 10 feet off the floor. Due to the weight of the flange assemblies, the work posed a safety hazard to personnel on the ground - especially when a flange was passed between the operator on the ladder and the operator on the ground. To alleviate the safety concern, the vent piping was extended from valve 4-940V to a point approximately 4 feet off the floor. The extension was accomplished using ½-inch welded stainless steel tubing and a new terminal vent isolation valve.

**Safety Evaluation:**

The changes implemented by this Engineering Package enhanced the ability to periodically vent the HHSI system piping. The vent path change did not adversely affect the operation, function, or design bases of the HHSI system. Additionally, no new failure modes were created by the passive piping and valve changes. The Engineering Package evaluated the new configuration and determined that the modified vent piping satisfied all of the applicable Updated FSAR loading conditions, including seismic loads. Since the response of the HHSI system during design basis accidents remained unchanged with the new vent piping design, the modification did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation

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## PLANT CHANGE/MODIFICATION 97-034

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 12/14/99 |

## APPENDIX R DOCUMENTATION CHANGES IN SUPPORT OF CRs 96-1072, 96-1432, 96-1440, AND 96-1452

**Summary:**

This Engineering Package revised the Turkey Point Units 3 and 4 Appendix R Safe Shutdown Analysis (SSA), Appendix R Essential Equipment List (EEL), Appendix R Essential Cable List (ECL), and Raceway Fire Protection Wrap drawings, to incorporate changes documented in Condition Report Nos. 96-1072, 96-1432, 96-1440, and 96-1452.

The revisions implemented by this Engineering Package involved changes in operator actions, requirements for raceway fire protection, and availability of components and equipment during postulated fire scenarios. Changes to the above documents were also made for miscellaneous cables and equipment associated with systems referenced in the above condition reports.

**Safety Evaluation:**

The document changes implemented by this Engineering Package were enveloped by established fire protection design criteria and regulatory requirements. In those cases where fire barrier requirements were removed by this Engineering Package, compensatory measures were identified or a justification provided which ensured continued availability of the safe shutdown function. The new proceduralized manual actions were evaluated to ensure that adequate time existed to perform them, and that adequate emergency (Appendix R) lighting and access and egress paths existed for successful completion. None of the normal and safe shutdown functions of equipment affected by this modification were altered. Based on the evaluation criteria provided in this Engineering Package, the changes did not constitute an unreviewed safety question, or require changes to the plant technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications

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## PLANT CHANGE/MODIFICATION 97-058

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 08/30/99 |

## MODIFICATIONS AND ADDITIONS TO AUTOMATIC SPRINKLER SYSTEMS

### Summary:

This Engineering Package modified the hydraulic configuration of the turbine building sprinkler system to accommodate a postulated turbine lube oil fire that results from a gross failure of the low-pressure turbine and/or generator bearing oil seals. The modifications included adding more sprinklers to protect safe shutdown circuits and equipment throughout the turbine building (including those protected with Thermo-Lag fire barrier material) and adding a down-turned pipe elbow and vortex breaker to the existing raw water tank (RWT-1) suction nozzle. The suction nozzle modifications were necessary to gain access to the currently unavailable water inventory located below the existing suction point such that the increased fire water demand could be satisfied. The combination of these changes provided the additional protection needed to accommodate the new fire scenario.

The postulated fire scenario and commitments for protection coverage are described in the exemption request submitted to the NRC via letter L-97-181.

### Safety Evaluation:

The modifications addressed by this Engineering Package did not impact the operation, function, or design basis of any safety related equipment. Increasing the degree of coverage provided by the existing fixed water spray system enhanced the ability to deliver the required water flow and pressure to the turbine building service points. No changes were made to any safe shutdown circuits or equipment that would alter the plant response to a postulated fire event. Additionally, the fire piping upgrades did not increase the probability of an internal flooding event. Since no new equipment or Operator actions were invoked by the changes, the modifications did not constitute an unreviewed safety question, or require a change to the plant technical specifications

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## PLANT CHANGE/MODIFICATION 98-017

| | | |
|---|---|---|
| UNIT | : | 3 |
| TURN OVER DATE | : | 03/22/00 |

### REPLACEMENT OF CONTAINMENT PURGE VALVE ACTUATORS (POV-3-2600 & 2601)

**Summary:**

This Engineering Package provided the engineering and design necessary to replace the existing containment purge supply valve actuators with new actuators to improve reliability, operability and maintenance. Valve POV-3-2600 provides the outside containment isolation barrier for the containment purge supply penetration. Valve POV-3-2601 provides the inside containment isolation barrier for the containment purge supply penetration. The existing actuators for these valves required frequent maintenance, and spare parts were not readily available to service them because they were obsolete. The existing actuators utilized a dual spring / air canister design whereas the new actuators utilize a single spring / air canister design. The new actuators provide more torque and require less instrument air pressure than the existing models to achieve an opening and closing stroke.

The containment purge return valve actuators POV-3-2602 and POV-3-2603 are manufactured by a different vendor and are not included in the replacement activity.

**Safety Evaluation:**

The existing containment purge supply valves POV-3-2600 and POV-3-2601 were upgraded in this design package to improve reliability, operability, and maintenance. The activity was considered to be a design enhancement since the new actuators were similar in form, fit and function to the existing actuators. No new failure modes were created as a result of the component replacement. Additionally, the modification did not adversely affect the integrity, operation or function of any safety related system. Since no functional changes were made, this modification did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of this modification.

## PLANT CHANGE/MODIFICATION 98-031

| | | |
|---|---|---|
| UNIT | : | 4 |
| TURN OVER DATE | : | 12/02/99 |

## INSTALLATION OF DRAIN LINE FOR UNIT 4 TRANSFER CANAL LEAK CHASE SYSTEM

### Summary:

This Engineering Package installed a drain line on the Unit 4 spent fuel pool (SFP) transfer canal leak chase system to improve the disposal of any water that leaks past the canal liner during fuel transfer operations. The stainless steel canal liner provides a leakage barrier to contain the borated water used to cool and shield the spent fuel when it is transferred between the SFP and the reactor cavity. Any water that may leak past the canal liner is collected in the transfer canal leak chase system (monitoring trenches). The transfer canal leak chase system is currently connected to the SFP leak chase system which is provided with a drain line. The existing drainage scheme is very slow and leaves water trapped behind the transfer canal liner plate after the transfer canal is emptied following core reload. This trapped water is at a slightly elevated pressure due to the static head of the borated water maintained in the transfer canal during refueling operations. This trapped water can cause the liner to bulge when the canal is emptied which can degrade the liner and underlying concrete, and potentially interfere with operation of fuel handling equipment. To alleviate this condition, a separate drain line and isolation valve was installed for the transfer canal leak chase system. The drain line was routed to the adjacent new fuel storage room. Use of the drain requires that the isolation valve be opened and a temporary hose connected between the drain pipe outlet and the new fuel storage room floor drain.

### Safety Evaluation:

The modifications addressed by this Engineering Package enhanced the drainage capability of the transfer canal leak chase system. No new failure modes were created by the new valve and drain line installation. The new piping was evaluated in accordance with seismic criteria contained in the FSAR to ensure that the installation did not create the possibility of any adverse interactions with safety related structures, systems and components. Based on the evaluation criteria contained in the Engineering Package, the modifications did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 98-032

| | | |
|---|---|---|
| UNIT | : | 3 |
| TURN OVER DATE | : | 03/27/00 |

## TURBINE SERVO-MOTOR TEST VALVE MODIFICATION

### Summary:

This Engineering Package modified the configuration of the control oil leakoff piping and valves downstream of the #1 and #3 turbine governor valve actuators to improve the operators ability to transfer turbine load from the left side governor valves to the right side governor valves during turbine stop valve testing. The turbine stop valves are periodically tested on-line at a power level of about 40%. The governor valves are used to manually control steam flow to the turbine during the test in order to maintain the desired turbine load at the generator synchronized speed. The existing control oil design utilized a separate motor-operated test/throttle valve and a separate leakoff line for each governor valve to regulate control oil backpressure during the test. A common control switch in the control room linked the two test valves together so that the #1 and #3 governor valves moved in unison during the load transfer operation; consistent with their normal control action. The modified design replaced the individual test valves with a common (motor-operated) pressure control valve such that the backpressure control function was linked both mechanically and electrically to a single control device. This change improved leakoff line flow control, enhanced the synchronized movement of the #1 & #3 governor valves during the test, and reduced the number of components that were required to operate during the test. Needle valves were also installed in the leakoff lines to permit each governor valve to be adjusted individually. A needle valve was similarly installed in the bypass line around the PCV to provide additional flexibility.

This Engineering Package also provided the necessary justification to remove the #2 and #4 test valves from the control oil system since they are no longer used for stop valve testing.

### Safety Evaluation:

The modifications performed by this Engineering Package did not adversely affect operation of the various turbine overspeed protection devices, or the auto-stop or control oil systems. Failure modes associated with the revised leakoff design were reviewed to ensure that the plant would not be placed in an unanalyzed condition during the turbine valve test. One new failure mode was identified in the evaluation and it pertained to a failure of the PCV diaphragm assembly. The effects of this additional failure mode was determined to be bounded by other passive failures including failure of the control oil piping. Consequently, the modifications described in this Engineering Package did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 98-039

UNIT : 4
TURN OVER DATE : 10/19/00

## PENETRATION 23 OVERPRESSURE PROTECTION

### Summary:

This Engineering Package redesigned the overpressure protection feature that was previously provided for containment penetration No. 23 to address Generic Letter 96-06 concerns. Generic Letter 96-06 was issued by the NRC to address the potential for thermally induced overpressurization of isolated water-filled piping, such as between containment barrier isolation valves. This condition has the potential to jeopardize the ability of systems to perform their safety related functions and could lead to a loss of containment integrity.

Containment penetration No. 23 is for a process line that allows removal of liquid waste collected in the containment sump during normal plant operation. The modification originally provided for this penetration consisted of a drilled hole in the discs of check valves 4-4692A and B. Upon implementation of that change (via PC/M 97-012), it was found that back flow through the drilled discs immediately after the sump pump shutdown was initiating a false alarm of high leakage rate inside Containment. To rectify this condition, this Engineering Package provided the necessary design documentation to replace the existing check valves with equivalent check valves without drilled discs and install a thermal relief valve on the affected section of pipe for overpressure protection. The new thermal relief valve was set to relieve at a pressure 10% higher than the lowest rated component within the isolated bounds. Additionally, a test connection and an isolation valve was provided on the discharge line of the containment sump pumps to facilitate local leak rate testing of penetration No. 23.

### Safety Evaluation:

The modifications addressed by this Engineering Package eliminated a potential failure mode for penetration No. 23. The provision of a thermal relief path for the affected piping segments did not alter any of the critical functional characteristics of the piping system. The affected piping segments were evaluated in accordance with seismic criteria contained in the FSAR to ensure that the installation did not create the possibility of any adverse interactions with safety related structures, systems and components. The UFSAR commitment that containment isolation be established assuming an independent single active failure remains intact with the modified design. Accordingly, the implemented changes did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 98-047

| | | |
|---|---|---|
| UNIT | : | 4 |
| TURN OVER DATE | : | 04/12/99 |

## TURBINE SERVO-MOTOR TEST VALVE REPLACEMENT

### Summary:

This Engineering Package modified the configuration of the control oil leakoff piping and valves downstream of the #1 and #3 turbine governor valve actuators to improve the operators ability to transfer turbine load from the left side governor valves to the right side governor valves during turbine stop valve testing. The turbine stop valves are periodically tested on-line at a power level of about 40%. The governor valves are used to manually control steam flow to the turbine during the test in order to maintain the desired turbine load at the generator synchronized speed. The existing control oil design utilized a separate motor-operated test/throttle valve and a separate leakoff line for each governor valve to regulate control oil backpressure during the test. A common control switch in the control room linked the two test valves together so that the #1 and #3 governor valves moved in unison during the load transfer operation; consistent with their normal control action. The modified design replaced the individual test valves with a common (motor-operated) pressure control valve such that the backpressure control function was linked both mechanically and electrically to a single control device. This change improved leakoff line flow control, enhanced the synchronized movement of the #1 & #3 governor valves during the test, and reduced the number of components that were required to operate during the test. Needle valves were also installed in the leakoff lines to permit each governor valve to be adjusted individually. A needle valve was similarly installed in the bypass line around the PCV to provide additional flexibility.

This Engineering Package also provided the necessary justification to remove the #2 and #4 test valves from the control oil system since they are no longer used for stop valve testing.

### Safety Evaluation:

The modifications performed by this Engineering Package did not adversely affect operation of the various turbine overspeed protection devices, or the auto-stop or control oil systems. Failure modes associated with the revised leakoff design were reviewed to ensure that the plant would not be placed in an unanalyzed condition during the turbine valve test. One new failure mode was identified in the evaluation and it pertained to a failure of the PCV diaphragm assembly. The effects of this additional failure mode was determined to be bounded by other passive failures including failure of the control oil piping. Consequently, the modifications described in this Engineering Package did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 98-051

| | | |
|---|---|---|
| UNIT | : | 4 |
| TURN OVER DATE | : | 04/19/00 |

## INTAKE STRUCTURE BAY WALLS CATHODIC PROTECTION

**Summary:**

Testing and inspection at the Intake Structure revealed the presence of corrosion damage at several of the steel reinforcing bars embedded in the bay walls. In order to preclude any further corrosion activity and to ensure that the intake structure remains within its design basis, this Engineering Package provided for the installation of an impressed current cathodic protection system at the intake structure bay walls and travelling screen housing support beams in all four of the Unit 4 intake wells.

Cathodic protection was provided for the north, south and east bay walls from the water line to the concrete deck of the intake structure, and in the west wall from the water line to the support deck for the circulating water pipe. The system also provided corrosion control for the reinforcing steel in the two concrete beams supporting the travelling screen housing.

**Safety Evaluation:**

The function of the installed cathodic protection system is to protect the intake structure bay walls and travelling screen housing support beams from the long-term effects of corrosion. It is intended to supplement existing methods of corrosion control at the intake structure and performs no safety related functions. The installation and final configuration of this modification did not create the possibility of any adverse interactions with existing safety related structures, systems, or components. Consequently, the modification in this Engineering Package did not constitute an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 99-019

| | | |
|---|---|---|
| UNIT | : | 3 |
| TURN OVER DATE | : | 03/23/00 |

## MOV ENHANCEMENT
## LIMITORQUE TECHNICAL UPDATE 98-01

### Summary:

This Engineering Package modified several safety related motor-operated valves (MOVs) to address a change in the application factor published in Limitorque Technical Update 98-01. The application factor is part of the equation used to size a MOV actuator. It is a dimensionless number specified by the actuator vendor which accounts for motor manufacturing variations not accounted for by the motor manufacturer. Technical Update 98-01 reduced the application factor from 1.0 to 0.9. Changing the application factor can affect the calculated available output torque of an actuator. To ensure that the safety related MOVs will be capable of accomplishing their safety functions with a reduced output torque, this Engineering Package changed the gear ratio on the actuators of valves MOV-3-863A & B, MOV-3-864A & B, MOV-3-872, MOV-3-843A & B, MOV-3-880A & B, MOV-878A, MOV-3-744A & B, MOV-3-1420, and MOV-3-1421. It also replaced the motors on the actuators of valves MOV-3-1417, MOV-3-1418, and MOV-3-6386 with larger capacity motors. The motor overloads and breakers were also upgraded accordingly.

Revision 1 was issued to change the control logic for valve MOV-3-716A so that it closed on the limit switch signal in lieu of the torque switch signal. This change was implemented to reduce the operating loads experienced by the MOV in its existing configuration.

### Safety Evaluation:

The modifications addressed by this Engineering Package increased the available output torque of several safety related MOVs to ensure that they will be capable of accomplishing their safety function under reduced voltage conditions (given the change in application factor). The resulting changes in valve stroke time were evaluated to ensure that existing safety analysis assumptions remained valid. In each case, it was demonstrated that the affected safety system would continue to function within analyzed bounds. Additionally, an engineering review demonstrated that the seismic qualification of the affected valves was not adversely affected by the modifications due to the small weight changes involved. Replacement gears were shown to be consistent with the original gear material, and compatible with the actuator materials of construction. Since the implemented changes did not alter any valve functions, or methods of valve actuation, the modifications implemented by this Engineering Package did not have any adverse effects on plant safety or operation. Consequently, the modifications did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-020

| | | |
|---|---|---|
| UNIT | : | 3 |
| TURN OVER DATE | : | 03/22/00 |

## MSIV CONTROL CIRCUIT LOGIC CHANGE

**Summary:**

This Engineering Package modified the Unit 3 main steam isolation valves (MSIVs) to prevent inadvertent closure of the valves due to a failure of an auxiliary control relay in the actuation logic. An auxiliary control relay is provided in each MSIV circuit to seal-in a manual or automatic MSIV closure signal. It is designed to maintain the MSIV in a closed position until the operator resets the seal-in condition by taking the respective MSIV control switch to the open position. The auxiliary relay was originally designed to be normally energized and to de-energize to provide the seal-in function. With this arrangement, any failure of the relay would cause the associated MSIV to close. Inadvertent closure of an MSIV at power will result in a reactor trip condition. This Engineering Package reversed the auxiliary control relay logic so that it will be de-energized during normal plant operation and energized when required to accomplish its intended function.

Failure of a MSIV auxiliary control relay has previously occurred on Unit 3 causing inadvertent closure of the affected MSIV and a Unit 3 reactor trip event. This design change was implemented to prevent a similar trip condition from occurring in the future.

**Safety Evaluation:**

The modification performed by this Engineering Package was evaluated to ensure that no new failure modes were created. It was concluded that the circuit changes did not alter the method of isolating the MSIVs during an accident, or retard the valve closing speed. The evaluation demonstrated that sufficient redundancy and electrical separation were retained in the modified design to ensure that the MSIVs will close as required under single active failure conditions. Since the design basis for main steam isolation was not affected by the relay logic change, the circuit modifications did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-045

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 02/23/00 |

## ATMOSPHERIC STEAM DUMP VALVE AIR / NITROGEN SUPPLY ENHANCEMENTS

**Summary:**

This Engineering Package modified the Unit 3 and 4 atmospheric steam dump valves (CV-*-1606, CV-*-1607, and CV-*-1608) to improve their operating performance. Sluggish operation of these valves was reported during post trip responses and design changes were implemented to improve the actuating air and nitrogen supplies to the valves. The high/low selectors in the air/nitrogen flowpaths were replaced with a check valve supply arrangement, and the existing flow regulators were replaced with higher flow models. Additionally, the standby response settings of the Hand/Auto Stations for control of the valves were changed from 1005 psig to 1000 psig steam pressure. This setpoint change was primarily made to improve the man-machine interface for steam dump system since a set pressure of 1000 psig falls directly on a scale division and does not require interpolation by Operators personnel.

**Safety Evaluation:**

The changes implemented by this Engineering Package enhanced the performance capability of the atmospheric steam dump valves (ADVs). The changes in the actuating air/nitrogen supply piping did not adversely affect the operation, function, or design bases of the main steam system. Additionally, no new failure modes were created by the passive piping and valve changes. Based on the evaluation criteria provided in this Engineering Package, the modification did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-047

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| UNIT | : | 3 |
| TURN OVER DATE | : | 03/28/00 |

### IN-MAST SIPPING, ULTRASONIC INSPECTION, AND FUEL ROD RECONSTITUTION

**Summary:**

This Engineering Package modified the reactor refueling machine to permit in-mast fuel sipping during the Unit 3 Cycle 18 refueling outage. Fuel sipping was required because routine chemistry samples of the primary coolant prior to the outage identified both an iodine and cesium spike via isotopic analysis – indicating the presence of one or more leaking fuel assemblies. Several modifications to the refueling machine mast were required to enable fuel sipping during the core offload. The modifications included installation of an air injection manifold at the bottom of the fixed mast assembly, installation of air supply tubing on the outside of the stationary mast to connect the air injection manifold to the air supply source, installation of an air collection manifold at the top of the mast, installation of covers to seal off the various openings in the fixed mast, and installation of an air suction system to transfer air from the collection manifold to the monitoring equipment.

The air supply and collection manifolds were considered to be temporary modifications to the refueling machine and were required to be removed upon completion of core reload activities.

This Engineering Package also provided for the temporary installation of support fixtures in the Unit 3 spent fuel pool (SFP) to permit fuel assembly ultrasonic inspection and fuel assembly reconstitution.

**Safety Evaluation:**

The majority of this modification was implemented during the Unit 3 Cycle 18 refueling outage. Potential failure modes resulting from the hardware installations and operation of the refueling machine over the open reactor core were reviewed to determine their impact on plant safety. The proposed permanent and temporary modifications had no functional or spatial interactions with any equipment important to safety. Additionally, no other new failure modes were created by the proposed modification, and the probability of occurrence and consequences of previously analyzed failures were not increased. Based on the evaluation criteria contained in the Engineering Package, the modifications 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 plant technical specifications. Accordingly, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-053

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| UNIT | : | 4 |
| TURN OVER DATE | : | 10/20/00 |

## REACTOR COOLANT PUMP "4C" MOTOR REFURBISHMENT AND UPGRADE

**Summary:**

This Engineering Package provided for the refurbishment and upgrade of the 4C reactor coolant pump (RCP) motor. The original installed motor was replaced with a spare motor which was refurbished at the Westinghouse Electro-Mechanical Division facility. This refurbishment consisted of inspection and maintenance activities performed to the existing design specifications. In addition, a multiport drain sump and labyrinth entry vent port were installed concurrent with the refurbishment to ensure consistency with the latest RCP technology, and to realize additional reliability and availability. The intent of this modification is to essentially eliminate the anomalous oil level alarms caused by dynamic fluid effects and improve oil pressure sensing characteristics of the RCP motor.

**Safety Evaluation:**

The only safety related function performed by the RCP motors is to maintain a sufficient amount of inertia (through its flywheel) to satisfy the coastdown flow requirements assumed in the plant safety analyses for protection against departure from nucleate boiling. The design bases established in the Updated FSAR were reviewed and determined not to be affected because the modifications met all Updated FSAR criteria stipulated for the original design. In addition, the modifications did not impact the hydraulic performance of the pump or the coastdown capability of the pump motor. Since the modifications performed by this Engineering Package did not have any adverse effect on plant safety or plant operations, the modifications did not constitute an unreviewed safety question or require changes to the plant technical specifications. Accordingly, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-056

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| UNIT | : | 4 |
| TURN OVER DATE | : | 10/23/00 |

## MOV ENHANCEMENT
## LIMITORQUE TECHNICAL UPDATE 98-01

### Summary:

This Engineering Package modified several safety related motor-operated valves (MOVs) to address a change in the application factor published in Limitorque Technical Update 98-01. The application factor is part of the equation used to size a MOV actuator. It is a dimensionless number specified by the actuator vendor which accounts for motor manufacturing variations not accounted for by the motor manufacturer. Technical Update 98-01 reduced the application factor from 1.0 to 0.9. Changing the application factor can affect the calculated available output torque of an actuator. To ensure that the safety related MOVs will be capable of accomplishing their safety functions with a reduced output torque, this Engineering Package changed the gear ratio on the actuators of valves MOV-4-880A & B, MOV-878B, MOV-4-744A & B, MOV-4-1420, and MOV-4-1421. It also replaced the motor on the actuator of valve MOV-4-6386 with a larger capacity motor. The breaker instantaneous trip setting was also increased accordingly.

Revision 1 was issued to change the control logic for valves MOV-4-716A & B so that they close on the limit switch signals in lieu of the torque switch signals. This change was implemented to reduce the operating loads experienced by the MOVs in their existing configuration.

### Safety Evaluation:

The modifications addressed by this Engineering Package increased the available output torque of several safety related MOVs to ensure that they will be capable of accomplishing their safety function under reduced voltage conditions (given the change in application factor). The resulting changes in valve stroke time were evaluated to ensure that existing safety analysis assumptions remained valid. In each case, it was demonstrated that the affected safety system would continue to function within analyzed bounds. Additionally, an engineering review demonstrated that the seismic qualification of the affected valves was not adversely affected by the modifications due to the small weight changes involved. Replacement gears were shown to be consistent with the original gear material, and compatible with the actuator materials of construction. Since the implemented changes did not alter any valve functions, or methods of valve actuation, the modifications implemented by this Engineering Package did not have any adverse effects on plant safety or operation. Consequently, the modifications did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 99-059

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| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 08/09/00 |

## FIRE PROTECTION - OUTDOOR
## ELECTRICAL RACEWAY FIREPROOFING REQUIREMENTS

**Summary:**

This Engineering Package revised the Turkey Point Units 3 and 4 Appendix R Safe Shutdown Analysis (SSA), Appendix R Essential Equipment List (EEL), Appendix R Essential Cable List (ECL), and Raceway Fire Protection Wrap drawings, to incorporate those changes necessary to reduce reliance on Thermo-Lag fire barrier material in certain outdoor fire areas. The required fire area changes were documented in a series of engineering evaluations that were prepared in response to Supplement 1 to NRC Bulletin No. 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function."

The revisions implemented by this Engineering Package involved changes in operator actions and/or the use of protected redundant components and equipment during postulated fire scenarios. Changes to the above documents were also made for miscellaneous cables and equipment associated with past plant modifications.

**Safety Evaluation:**

The document changes implemented by this Engineering Package were enveloped by established fire protection design criteria and regulatory requirements. In those cases where fire barrier requirements were removed by this Engineering Package, compensatory measures were identified or the use of protected redundant equipment was specified to ensure continued availability of the safe shutdown function. The new proceduralized manual actions were evaluated to ensure that adequate time existed to perform them, and that adequate emergency (Appendix R) lighting and access and egress paths existed for successful completion. None of the normal and safe shutdown functions of equipment affected by this modification were altered. Based on the evaluation criteria provided in this Engineering Package, the changes did not constitute an unreviewed safety question, or require changes to the plant technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 00-008

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| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 09/18/00 |

### FIRE PROTECTION – SERVICE WATER ALTERNATIVES TO THE JOCKEY PUMPS

**Summary:**

This Engineering Package provides a cross-tie between the plant service water system discharge header and the plant fire water main. The cross-tie will allow the fire main to be maintained water-solid and pressurized using the service water system water supply. In the event that service water is not available, the jockey pumps can be put back in service to maintain the fire loop in its standby condition. Utilizing the service water alternative is advantageous because the makeup and pressure control function is completely passive in nature, and its use will increase the service life and reduce maintenance costs for the jockey pumps.

Installation of the cross-tie feature required the addition of small bore piping, valves, restriction orifices, and a flow indicator. The cross-tie is sized to make up lost volume from small leaks while allowing some waterflow alarm tests, but is also designed to allow the fire pump to start in the event of a fire emergency. The design basis for the jockey pumps was considered in establishing the capacity of the cross-tie.

This modification does not affect fire water supply or delivery. In the event a fire suppression system operates, only fire protection equipment (fire pumps, fire piping, dedicated water) are utilized.

**Safety Evaluation:**

The modification addressed by this Engineering Package enhanced the operation of the fire protection system. The changes did not alter the operation, function, or design basis of any structure, system, or component considered important to safety. Additionally, the piping modification did not introduce any new functional or spatial interactions with equipment considered important to safety. Since no new equipment or Operator actions were invoked by the changes, the modifications did not constitute an unreviewed safety question, or require a change to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 00-009

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| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 03/28/00 |

## STEAM GENERATOR FLEXIBLE TUBE STAKES

**Summary:**

This Engineering Package provided for the installation of ABB Combustion Engineering designed steam generator flexible tube stabilizers (stakes) and plugs in the Turkey Point Units 3 and 4 steam generators. The flexible tube stakes were required to be installed in those steam generator tubes with circumferential indications near the expansion transition region, i.e., near the top of the tubesheet. The flexible tube stakes function to restrain severed tubes and dampen vibration to mitigate additional wear on plugged tubes. Tube stakes are designed such that if a staked tube were to become fully severed at the secondary side of its tubesheet, it would prevent damage to adjacent tubes due to excessive vibration. This change updated plant documents to allow the use of tube stabilizers for plugged tubes, and identified tube plug designs to be used with the stabilizer.

Attachment 4.1 to this Engineering Package provided the recommended FSAR changes to document that defective steam generator tubes having indications may require corrective maintenance actions such as plugging or plugging and staking.

**Safety Evaluation:**

The modification addressed by this Engineering Package was evaluated to ensure that no new failure modes were created. It was concluded that the tube stabilizer was a structural support component for the steam generator since its function was to support or stabilize a degraded steam generator tube. Failure of a steam generator support component is not postulated in the Turkey point FSAR. Since no new hazards are created by the installation of flexible tube stabilizers in the steam generators, the actions and documentation changes identified in this Engineering Package did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 00-019

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| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 08/30/00 |

## CONTAINMENT ISOLATION AIR SAMPLE VALVES INDICATION MODIFICATION

**Summary:**

This Engineering Package provided the necessary design documentation to re-wire the limit switches and associated relays for isolation valves SV-*-2911, SV-*-2912, and SV-*-2913, associated with the containment atmosphere gaseous and particulate radiation monitors. The wiring changes were required to correct position indication errors. The existing wiring scheme provided 'open / not open' indication instead of the 'open / closed' indication specified in Tables 7.5-1 and 7.5-2 of the FSAR. To resolve the discrepancy, the valve indication circuit was re-wired to provide 'closed / not closed' indication. Although the installed equipment could not be configured to provide the FSAR required position indication, the 'closed / not closed' indication was considered to be an enhancement since it satisfied the position indication requirement of Regulatory Guide 1.97, without exception. The containment isolation function of these valves was not affected by the proposed circuit changes.

An FSAR Change Package was provided as an attachment to this Engineering Package to document the conforming changes to Tables 7.5-1 and 7.5-2.

**Safety Evaluation:**

The modifications addressed by this Engineering Package corrected containment isolation valve position indication discrepancies and brought the plant into full compliance with Regulatory Guide 1.97. The wiring changes did not alter any valve functions or methods of valve actuation. Since the design basis for containment isolation was not affected by the wiring changes, the circuit changes did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the hardware and documentation changes.

## PLANT CHANGE/MODIFICATION 00-023

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| UNIT | : | 4 |
| TURN OVER DATE | : | 10/20/00 |

## MOV-4-843A, MOV-4-843B AND MOV-4-869 EQUALIZING LINES AND SIS MODIFICATIONS

### Summary:

This Engineering Package made several modifications to the plant to improve high head safety injection (HHSI) system venting and reduce the potential for gas binding of the HHSI pumps. The modifications included a) re-routing the existing bonnet equalizing lines for the HHSI system discharge isolation valves, MOV-4-843A and MOV-4-843B, from the HHSI side of the valves to the reactor coolant system (RCS) side of the valves, b) replacing the drilled disk on valve MOV-4-869 with a solid disk and a bonnet equalizing line to improve the isolation capability of the valve against back leakage during plant operation, c) installing a new check valve in each of the safety injection accumulator fill lines, and d) installing new high point vents in the HHSI pump discharge piping to the RCS hot and cold legs.

The equalizing line changes allow any nitrogen saturated fluid trapped in the bonnet cavities to be vented back to the RCS during depressurization events, and eliminates the potential for nitrogen intrusion into the HHSI system during plant operation. Connection to the RCS was made at existing drain valve locations.

All of the above changes were made outside containment.

### Safety Evaluation:

The modifications addressed by this Engineering Package eliminated a potential failure mode for the HHSI pumps. No new failure modes were created by the new valve and equalizing line installations. The new tubing and supports were evaluated in accordance with seismic criteria contained in the FSAR to ensure that the installation did not create the possibility of any adverse interactions with safety related structures, systems and components. Based on the evaluation criteria contained in the Engineering Package, the modifications did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 00-030

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| UNIT | : | 4 |
| TURN OVER DATE | : | 10/20/00 |

## UNIT 4 IN-MAST TELESCOPE FUEL SIPPING MODIFICATION

**Summary:**

This Engineering Package modified the reactor refueling machine to permit in-mast fuel sipping during the Unit 4 Cycle 19 refueling outage. Fuel sipping was required because routine chemistry samples of the primary coolant prior to the outage identified both an iodine and cesium spike via isotopic analysis – indicating the presence of one or more leaking fuel assemblies. Several modifications to the refueling machine mast were required to enable fuel sipping during the core offload. The modifications included installation of two suction nozzles on the fuel assembly gripper, two water hoses to connect the suction nozzles to the sipping diagnostic equipment, and a spring-loaded take-up reel directly above the refueling mast to eliminate slack during mast movement.

**Safety Evaluation:**

The majority of this modification was implemented during the Unit 4 Cycle 19 refueling outage. Potential failure modes resulting from the hardware installations and operation of the refueling machine over the open reactor core were reviewed to determine their impact on plant safety. The proposed modification had no functional or spatial interactions with any equipment important to safety. Additionally, no other new failure modes were created by the proposed modification, and the probability of occurrence and consequences of previously analyzed failures were not increased. Based on the evaluation criteria contained in the Engineering Package, the modifications 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 plant technical specifications. Accordingly, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 00-031

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| UNIT | : | 3 & 4 |
| TURN OVER DATE | : | 10/18/00 |

### REPLACEMENT OF EMERGENCY CONTAINMENT FILTER TYPE II CHARCOAL CELLS

**Summary:**

This Engineering Package provided the necessary design documentation to allow the use of replacement carbon cells in the emergency containment filters that are functionally equivalent to the original Turkey Point components, but differ from the FSAR description.

The replacement carbon cells were ordered in support of the wholesale replacement of all carbon cells during the Unit 4 Cycle 19 Refueling Outage. Some of the cells differ in their external configuration but meet all functional testing criteria and are dimensionally compatible with the emergency containment filter carbon cell rack.

An FSAR change package was provided as an attachment to this Engineering Package. The proposed change revises the physical description of the charcoal cells such that it is applicable to both the original and replacement cell designs. It also updates the information provided on charcoal type.

**Safety Evaluation:**

The modifications described in this Engineering Package did not adversely affect the design or function of the emergency containment filter system to reduce the iodine concentration in the containment atmosphere following a maximum hypothetical accident to levels assumed in the plant safety analysis. Since no new failure modes were created by the replacement carbon cells, the modifications implemented by this Engineering Package did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

# SECTION 2

# SAFETY EVALUATIONS

**SAFETY EVALUATION JPN-PTN-SENJ-88-052**
REVISION 4

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 03/15/00 |

## SAFETY EVALUATION FOR CONTAINMENT BULK TEMPERATURE


### Summary:

This safety evaluation was prepared to address the 10 CFR 50.59 criteria for operating the plant with containment temperature above the design basis limit for short periods of time during the hot summer months. No configuration or procedural changes were involved - only the temperature limit used as a basis for plant operation. The evaluation considered the impact of elevated ambient temperature conditions on structural integrity, cable ampacities, accident analyses, equipment qualification, and instrumentation accuracy. It was concluded that raising the containment bulk ambient temperature limit from 120°F to 125°F for a cumulative period of two weeks per year would be acceptable with no adverse impact on plant safety or operation.

Revision 4 evaluated the impact of shorter refueling outages on the environmental qualification of equipment inside containment. It was determined that the increased exposure to containment temperature conditions did not reduce the qualified life of the equipment below that assumed in the environmental qualification documentation packages, even at the elevated temperatures permitted during the summer months.


### Safety Evaluation:

This safety evaluation addressed the technical and licensing associated with operating the plant at a containment temperature of 125°F for a cumulative period of two weeks per year. It concluded that the safety related electrical equipment inside containment would continue to perform its specified safety functions with these short temperature excursions, even with the increased focus on shorter refueling outages. Since the proposed operating strategy was bounded by the technical specifications and did not change the analysis of accidents addressed in the FSAR, the specified temperature limitations did not constitute an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required to implement the actions identified in this safety evaluation.

## SAFETY EVALUATION JPN-PTN-SEEJ-89-085
REVISION 14

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| UNIT | : | 3 |
| APPROVAL DATE | : | 02/24/00 |

## DE-ENERGIZATION OF UNIT 3 4160 VOLT SAFETY RELATED BUSSES


### Summary:

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

Revision 14 updated the bus loading to coincide with recent design changes and changes in plant operating requirements.


### Safety Evaluation:

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

**SAFETY EVALUATION JPN-PTN-SECJ-95-001**
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 05/13/99

**SAFETY EVALUATION FOR SPECIFICATION SPEC-M-20**
**INSTALLATION OF "QUICK CONNECT COUPLING"**
**ON AIR OPERATED VALVES**

**Summary:**

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-M-020 in the maintenance process, in lieu of the current practice which required that a Plant Change/Modification (PC/M) package be issued and implemented for all such cases. It also demonstrated that the Specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-M-020 was developed to provide generic guidance for the installation of quick connects on air operated valves (AOVs) when required. These quick connects may be installed on instrument air lines, when routine maintenance or testing is performed on the valves in the Maintenance AOV Program. Each quick connect will require an Engineering review and approval prior to implementation. This review was intended to address all applicable loads, including the effects of vibration, spatial interactions, material selection, and configuration management requirements. Each Engineering review will be documented on a Maintenance Request Approval (MRA) form contained in Appendix A of the generic specification. This generic engineering specification was applicable to all plant safety classifications.

Revision 1 clarified that the piping class boundary of the installation is at the root isolation valve.

**Safety Evaluation:**

The safety evaluation concluded that the generic specification requirements and installation guidelines for quick connects would not have any adverse effects on safety related systems required to prevent or mitigate the consequences of design basis accidents. Similarly, the installation guidelines would not have any adverse effects on the Instrument Air System when used for its licensed functions during fires. The actions and guidance identified in the generic engineering specification and associated safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or guidance identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SENP-95-007

REVISION 4

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| UNIT | : | 3 |
| APPROVAL DATE | : | 03/29/00 |

## SAFETY EVALUATION FOR OPERABILITY OF RHR DURING INTEGRATED SAFEGUARDS TESTING


**Summary:**

This safety evaluation reviewed the Unit 3 engineered safeguards integrated test (ESIT) procedures with respect to a generic Westinghouse concern related to the effectiveness of the steam generators (S/Gs) to remove decay heat during shutdown conditions. Westinghouse identified that there was a potential for gas formation within the steam generator U-tubes under certain reactor coolant system (RCS) pressure and level conditions in Mode 5 that could inhibit the ability to establish natural circulation cooling. To accommodate the potential unavailability of the S/Gs for decay heat removal under these conditions, plant technical specifications require that both trains of the residual heat removal system (RHR) be operable in Mode 5 when the RCS is in a "loops not filled" configuration. Since safeguards testing was normally performed during Mode 5 with the RCS depressurized and partially drained, this evaluation was developed to document that both trains of the RHR system would remain operable during the test period. The evaluation concluded that no restrictions on plant operations or additional operator actions, other than those already prescribed in the ESIT procedures, were required to ensure RHR operability.

Revision 4 addressed the ability to divert excess component cooling water (CCW) flow through an inoperable CCW heat exchanger to maintain component flow rates within established limits during performance of the ESIT. Based on the decay heat load assumptions used in the evaluation, operation of the CCW system with ICW flow isolated to one of three heat exchangers was restricted to Mode 5 (post core reload) and Mode 6.


**Safety Evaluation:**

This safety evaluation examined the electrical, mechanical, and hydraulic configuration of the plant during performance of the ESIT in Modes 5 (loops not filled) and 6 (vessel level two feet below the flange) to ensure that both RHR loops would remain operable during the test sequence. It also evaluated CCW system operability with shell-side (CCW) flow being maintained through a heat exchanger with no corresponding tube-side (ICW) flow. Since all licensing and design basis requirements would continue to be met during the ESIT, the proposed changes did not involve an unreviewed safety question or require changes to plant technical specifications. Thus, prior NRC approval was not required to initiate the test sequences.

## SAFETY EVALUATION JPN-PTN-SENP-95-023
REVISION 4

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| UNIT | : | 4 |
| APPROVAL DATE | : | 10/13/00 |

## SAFETY EVALUATION FOR OPERABILITY OF RHR DURING INTEGRATED SAFEGUARDS TESTING

**Summary:**

This safety evaluation reviewed the Unit 4 engineered safeguards integrated test (ESIT) procedures with respect to a generic Westinghouse concern related to the effectiveness of the steam generators (S/Gs) to remove decay heat during shutdown conditions. Westinghouse identified that there was a potential for gas formation within the steam generator U-tubes under certain reactor coolant system (RCS) pressure and level conditions in Mode 5 that could inhibit the ability to establish natural circulation cooling. To accommodate the potential unavailability of the S/Gs for decay heat removal under these conditions, plant technical specifications require that both trains of the residual heat removal system (RHR) be operable in Mode 5 when the RCS is in a "loops not filled" configuration. Since safeguards testing was normally performed during Mode 5 with the RCS depressurized and partially drained, this evaluation was developed to document that both trains of the RHR system would remain operable during the test period. The evaluation concluded that no restrictions on plant operations or additional operator actions, other than those already prescribed in the ESIT procedures, were required to ensure RHR operability.

Revision 4 addressed the ability to divert excess component cooling water (CCW) flow through an inoperable CCW heat exchanger to maintain component flow rates within established limits during performance of the ESIT. Based on the decay heat load assumptions used in the evaluation, operation of the CCW system with ICW flow isolated to one of three heat exchangers was restricted to Mode 5 (post core reload) and Mode 6.

**Safety Evaluation:**

This safety evaluation examined the electrical, mechanical, and hydraulic configuration of the plant during performance of the ESIT in Modes 5 (loops not filled) and 6 (vessel level two feet below the flange) to ensure that both RHR loops would remain operable during the test sequence. It also evaluated CCW system operability with shell-side (CCW) flow being maintained through a heat exchanger with no corresponding tube-side (ICW) flow. Since all licensing and design basis requirements would continue to be met during the ESIT, the proposed changes did not involve an unreviewed safety question or require changes to plant technical specifications. Thus, prior NRC approval was not required to initiate the test sequences.

## SAFETY EVALUATION JPN-PTN-SEMS-96-003
REVISIONS 2 and 3

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| UNIT | : | | 4 |
| APPROVAL DATES | : | Rev. 2 | 04/12/99 |
| | | Rev. 3 | 06/10/99 |

## SAFETY EVALUATION FOR UNIT 4 STEAM GENERATORS' SECONDARY SIDE FOREIGN OBJECTS


**Summary:**

This evaluation addressed the potential safety significance of operating the Unit 4 steam generators (S/Gs) with irretrievable foreign objects present in the secondary side. Previously, individual safety evaluations addressed the acceptability of continued Unit 4 operation with foreign objects remaining in the S/Gs and associated systems. The purpose of this evaluation was to: (1) re-examine the analyses, results, requirements, and restrictions of previous evaluations while applying recent industry standards; (2) document the methodology for determining the interval between S/G eddy current tests as affected by estimated S/G tube wall wear times; and (3) provide a single Unit 4 safety evaluation to assess and document all the Unit 4 S/G foreign object estimated wear times as adjusted by updated S/G eddy current data and steam generator Foreign Object Search and Retrievals (FOSAR) results. FPL maintains a visual inspection program of the secondary side of S/Gs (in addition to the other inspection programs for S/Gs) to help prevent and detect the presence of loose parts.

Revision 2 incorporated results of the S/G inspections performed during the Cycle 18 refueling outage. Revision 3 modified the technique used to calculate wear time endpoints to be consistent with Revision 1 of WCAP-14258. It also provided additional analysis for the assumed foreign objects from the 4B feedwater pump previously evaluated in Revision 2.


**Safety Evaluation:**

Previous safety evaluations documented for each S/G secondary side foreign object have considered the effects of the object upon tube integrity, chemistry, S/G instrumentation, the main steam system, and S/G blowdown and sampling systems. This current evaluation established wear time to minimum tube wall thickness estimates based on conservative assumptions from Westinghouse WCAP-14258 and associated clarification correspondence. These wear times assume worst case conditions and actual wear times are likely to be much greater than the WCAP methodology would predict. Based on this assessment, this evaluation determined that currently identified foreign objects within the secondary side of the Unit 4 S/Gs did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the actions identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SEMS-96-014
REVISION 2

UNIT : 3 & 4
APPROVAL DATE : 08/27/99

## SAFETY EVALUATION FOR A TEST OF THE USE OF SUB-MICRON ULTRAFINE FILTERS IN THE CVCS AND SFP SYSTEMS

### Summary:

This evaluation served to allow the temporary use of the ultrafine cartridges with absolute filtration ratings in the reactor coolant system (RCS), seal water injection, and seal water return filters in the chemical volume and control system to reduce plant radiation levels and to extend the life of reactor coolant pump seals. The ultrafine filter program will proceed in three phases. Because the filters proposed for use must be specifically designed for the individual filter housings, Phase I will involve a demonstration for proper filter fit and performance of near equivalent rated absolute filters cartridges. Only one test cartridge will be installed in the parallel filter paths at a time; the other path(s) will contain rated filters of the type currently used. Phase II of the testing program is a gradual reduction in the absolute rating of the filters used. This will gradually filter out finer and finer particles as the overall RCS particulate inventory is reduced. This will continue until the desired RCS cleanliness level is reached. Phase III involves the permanent use of these filters under formal plant design change documentation. Phase I of the program was evaluated in a previous safety evaluation. This evaluation only addressed Phase II of the ultrafine filter program.

Revision 2 allowed the use of ultrafine filters for seal water injection, seal water return, RCS and spent fuel pool (SFP) filtration with the installation of a parallel, equal or larger pore size filter as backup.

### Safety Evaluation:

This evaluation addressed the use of ultrafine filter cartridges for the RCS, seal water return, seal water injection, and SFP filters. This evaluation concluded that these ultrafine filters will meet all current design criteria for the systems identified above. Failure modes were evaluated and precautions have been established to monitor these filters more closely during the test period. The use of these filters does not change system design bases, functions, and operation of any safety related equipment, and will not adversely affect any safety related structures, systems or components. Therefore, the testing, implementation and plant actions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-97-009
REVISION 0

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| UNIT | : | 3 |
| APPROVAL DATE | : | 04/11/00 |

## IN SITU HYDROSTATIC TESTING OF STEAM GENERATOR TUBE FLAWS

**Summary:**

This evaluation was prepared as a contingency measure in the event that in situ hydrostatic test of a steam generator (S/G) tube was required during the Turkey Point Unit 3 Cycle 18 refueling outage. Historically, pressure testing has been performed in the laboratory, thus requiring removal of the degraded tubes from the S/G to demonstrate that they can withstand the pressure requirements of draft Regulatory Guide (RG) 1.121, *"Basis for Plugging Degraded PWR Steam Generator Tubes."* More recently equipment has been developed to pressure test tubes in situ. Guidelines for in situ testing of S/G tubes have been developed by the industry and formally published by EPRI. They provide a guide to develop and justify plant specific in situ pressure test procedures as a means to assess the structural integrity of a degraded tube. One approach to validate the results of the examination techniques is to demonstrate via pressure testing that defective tubes can sustain the pressure requirements of draft RG without bursting or leaking beyond analyzed limits. This evaluation provides the pressures and plant restrictions/criteria for the in situ pressure testing, as well as the basis for the test pressures to be used and demonstrates that the test pressures meet the criteria of RG 1.121 and that they are safe to conduct in accordance with 10CFR50.59 requirements.

**Safety Evaluation:**

This evaluation examined the test equipment and the methods used to test the various types of flaw indications, the test pressure and hold time, the impact of testing on plant operation, failure modes and effects, plant operating restrictions during testing, and applicable compensatory measures. The evaluation concluded that the proposed testing approach and test equipment had no adverse impact on plant safety or plant operations, and therefore, did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to perform the subject pressure test.

## SAFETY EVALUATION PTN-ENG-SEES-97-094
REVISION 2

UNIT : 4
APPROVAL DATE : 07/20/00

## TEMPORARY INSTALLATION OF REMOTE MONITOR FOR 'C' RCP OIL LEVEL VERIFICATION


### Summary:

This safety evaluation addressed the temporary installation of a video transmitter, power supply, NEMA 4 enclosure, and approximately 240 feet of cable inside the Unit 4 containment to monitor the 4C reactor coolant pump (RCP) motor oil level. Due to the small weight involved (20 pounds) the equipment was secured to an existing steel support on the 30'-6" elevation of the containment. The video signal from the transmitter was routed to a communication box near the elevator platform on the 30'-6" elevation of the containment building, and connected to spare telephone leads which terminated outside containment in the cable spreading room.

Revision 2 evaluated the option of installing the power supply and associated cabling in the cable spreading room to allow the camera to be turned off as desired between inspection intervals. The camera configuration previous evaluated left the power on continuously which degraded the camera lens.


### Safety Evaluation:

An engineering review demonstrated that the equipment would remain in place during a design basis seismic event, and not damage adjacent equipment considered important to safety. It also demonstrated that the containment hydrogen, free volume, heat sink, and combustible loading analyses would not be adversely affected by the additional equipment (including the video transmitter and power supply) due to the small mass of material involved. Since the installation of video transmitter equipment and associated cabling did not change the operation, function, or design basis of any structure, system, or component important to safety, the actions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-98-048
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 12/14/99 |

## SAFETY EVALUATION FOR UFSAR CHANGES TO REFLECT TURBINE BUILDING SPRINKLER UPGRADES

**Summary:**

This safety evaluation was prepared to update the Turkey Point FSAR and Design Basis Documents (DBDs) to a) reflect the turbine building sprinkler modifications performed under PC/M 97-058, and b) incorporate new fire protection specification requirements.

The turbine building sprinkler system was enhanced by PC/M 97-058 to protect additional safe shutdown circuits from a postulated turbine lube oil fire. The increased demand for spray water created by the additional sprinklers was accommodated by modifying the Raw Water Tank 1 fire pump suction piping to increase the volume dedicated to fire protection.

The new fire protection specification requirements established in the evaluation ensure protection of safe shutdown capability by assuring operability of the turbine building sprinkler systems during plant operation.

Attachment 1 to the evaluation provided the recommended FSAR changes. Attachment 2 provided the recommended DBD changes.

**Safety Evaluation:**

The proposed FSAR and DBD changes summarized the additional fire suppression capabilities provided via PC/M 97-058, and provided new operability and surveillance requirements for those capabilities. The document updates did not alter the postulated fire scenario or the equipment required to mitigate its consequences. Consequently, the document changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the changes identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SECS-98-058
REVISION 1

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| UNIT | : | 3 |
| APPROVAL DATE | : | 08/10/00 |

## STORAGE OF TOOLS AND EQUIPMENT IN CONTAINMENT DURING ALL MODES OF OPERATION


**Summary:**

This evaluation addressed the acceptability of leaving a quantity of tools and equipment within the Unit 3 containment structure during all modes of plant operation. The items to be stored, and the storage locations within the Unit 3 containment, were specifically identified within the evaluation. The purpose of leaving these tools and equipment within containment following refueling outages was to reduce the usage demand on the Unit 3 polar crane during refueling outages. This evaluation considered the potential for adverse seismic interactions with safety related equipment, the potential for additional hydrogen generation within containment during accidents, the impact on the containment free volume and heat sink analyses, the potential to obstruct flow to the containment sumps, and the impact on containment combustible loading. To ensure that the tools and equipment addressed in the evaluation were safely stored during plant operation, both generic and specific actions and restrictions were identified for implementation within the evaluation.

Revision 1 evaluated the additional impact of leaving two scaffold platforms, two rigging beams, and a tool box in containment during repair of the 3D normal containment cooler. These additional items were required to be removed from containment upon completion of the maintenance work.


**Safety Evaluation:**

The safety evaluation concluded that the proposed items identified within the safety evaluation can safely remain within containment provided that all the restrictions and requirements identified within the evaluation were implemented. The evaluation further concluded that the identified restrictions and requirements would ensure that these activities would have no adverse effects on plant operation, and would not compromise the safety and licensing bases for Unit 3. Consequently, the requirements and restrictions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the requirements or restrictions identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SEMS-99-001**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 07/27/99 |

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 4

### Summary:

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143 dated July 1, 1997.

FSAR Chapter 4 provides a basic overview of the reactor coolant system design and operation. The review of Chapter 4 identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

Revision 1 of this safety evaluation incorporated minor changes made as a result of comments received during a Plant Nuclear Safety Committee meeting. All conclusions of Revision 0 remain unaffected as a result of this revision.

### Safety Evaluation:

This review has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not constitute an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION PTN-ENG-SEMS-99-003**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 8/24/99 |

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 9

### Summary:

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143 dated July 1, 1997.

FSAR Chapter 9 provides a basic overview of the chemical and volume control system, auxiliary coolant system, sampling system, fuel handling system, facility services, equipment and system decontamination, auxiliary building ventilation and containment purge, and post-accident hydrogen control.

The activity being performed by this evaluation updates the documentation in the FSAR to accurately reflect the facility in terms of how it is operated and what equipment is present in the field. No physical changes are being made to the facility or its manner of operation. These documentation changes have been evaluated and do not identify cases where the field condition is inappropriate or in conflict with the plant design bases.

### Safety Evaluation:

This evaluation has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not involve an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION PTN-ENG-SEMS-99-004**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 5/13/99 |

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 10

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143 dated July 1, 1997.

FSAR Chapter 10 provides a basic overview of the steam and power conversion system. The review of Chapter 10 identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

**Safety Evaluation:**

This review has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not constitute an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION PTN-ENG-SENS-99-008**
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 10/13/00

## SAFETY EVALUATION FOR CONDUCTING RCS FILL AND VENT ACTIVITIES DURING ENGINEERED SAFEGUARDS INTEGRATED TESTING


**Summary:**

The Engineered Safeguards Integrated Test (ESIT) is performed at the end of each refueling outage to demonstrate that the accident mitigating equipment is functioning properly prior to a plant startup. Over the past several years, it has been general practice to perform the test early in the post-refueling startup sequence while the RCS is depressurized in a "loops not filled" condition. Startup activities such as fill and vent would typically be performed after successful completion of the ESIT and generally takes 2 – 3 shifts to complete. In an effort to improve the post-refueling startup schedule, this safety evaluation looked at performing RCS fill and vent activities during, or prior to, the safeguards test. It examined the impact of performing the ESIT in Mode 5 with the RCS pressurized. It also examined the command and control aspects associated with integrating the two major startup evolutions together.

Each system and component utilized in the fill and vent process was reviewed to ensure that the required equipment would be available during the various safeguards tests. Appropriate operating restrictions were established to prevent the RCS from exceeding the overpressure mitigating system actuation setpoint when the various pumps connected to the RCS start and stop under the simulated accident signals with water solid conditions in the pressurizer.

The evaluation demonstrated that integrating RCS fill and vent activities with the post-refueling safeguards test did not introduce any new failure modes for the RCS or its support systems. The RCS would continue to be operated within analyzed limits.

The original evaluation was prepared for Unit 4. Revision 1 extended the scope of that evaluation to include Unit 3.


**Safety Evaluation:**

The safety evaluation demonstrated that RCS fill and vent activities can be performed during those windows of opportunity when actual safeguards train testing is not in progress and that the integration of the two activities did not involve an unreviewed safety question, or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the ESIT procedure changes.

## SAFETY EVALUATION PTN-ENG-SEMS-99-010
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 09/14/00

## SAFETY EVALUATION FOR RCS CHEMICAL DEGASSING

**Summary:**

This safety evaluation analyzed the impact on plant safety and operation associated with using hydrogen peroxide to chemically remove dissolved hydrogen from the reactor coolant during plant cooldowns. Hydrogen peroxide is routinely used in the reactor coolant system (RCS) at cold shutdown conditions to provide a controlled solubilization of radio-cobalt for subsequent removal via the chemical and volume control system (CVCS) demineralizers. Removal of radio-cobalt limits radiation exposure from reactor coolant borne radio-cobalt sources when the plant is shut down. The proposed chemical degassing process extends the hydrogen peroxide treatment to provide for reactor coolant dissolved hydrogen removal. Industry experience has demonstrated that hydrogen peroxide will react rapidly with dissolved hydrogen in cold borated coolant, in near stoichiometric proportions, with pure (unborated) water as the product. The process enables RCS degassing to be completed in parallel with plant cooldown. The evaluation addressed: a) the potential to form flammable mixtures in the RCS and CVCS gas spaces, (b) the impact on core reactivity caused by the reaction product (pure water), and c) the potential to increase process instrument corrosion.

Revision 1 enabled the RCS degassing and radio-cobalt removal to be performed simultaneously rather than sequentially during cooldown.

**Safety Evaluation:**

This safety evaluation defined the necessary plant configuration, precautions and method of adding hydrogen peroxide to the RCS that will ensure safe and efficient chemical degassing. It demonstrated that the addition of hydrogen peroxide to accomplish the degassing function did not affect any assumptions relative to accident initiators, did not impede the accomplishment of post-accident recovery efforts, or increase the consequences of postulated accidents. Since plant design requirements continued to be met and the integrity of the reactor coolant system pressure boundary was not challenged, it was concluded that the actions or plant procedure changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the alternate RCS degassing procedure.

## SAFETY EVALUATION PTN-ENG-SENS-99-011
REVISION 1

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 04/15/99 |

## SAFETY EVALUATION RELATED TO AREA DESIGNATION OF THE COOLING CANALS

### Summary:

This evaluation was developed to address concerns relating the appropriate definition of area designation for the cooling canals consistent with regulatory requirements and definitions. In 1997 it was determined that the cooling canals were not being controlled as "Restricted Areas" as defined in the FSAR and in 10 CFR 20. Corrective actions were taken to provide additional training to personnel and to post the area appropriately to comply with the requirements of a Restricted Area. Further review of this issue determined that the Turkey Point cooling canals were never clearly documented as a 10 CFR 20 Restricted Area in licensing documentation and that the area could be viewed as a "Controlled Area" as defined in NUREG/CR-6204.

An FSAR Change Package was provided as an attachment to this safety evaluation. The FSAR Change Package identified the necessary changes to the FSAR text and provided an updated drawing of the "controlled area" boundary of the cooling canal system.

Revision 1 of this safety evaluation incorporated minor changes made as a result of comments received during a Plant Nuclear Safety Committee meeting.

### Safety Evaluation:

The safety evaluation determined that changing the designation of the cooling canals to a "controlled area" was an administrative activity. The evaluated changes to the Turkey Point FSAR for correctness and clarification did not change the operation, function, or design bases of any structure, system, or component important to safety. Consequently, the FSAR changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Thus, prior NRC approval was not required to implement the documentation changes.

**SAFETY EVALUATION PTN-ENG-SEFJ-99-013**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 10/28/99 |

## SAFETY EVALUATION FOR THE USE OF WABA FUNNEL DEVICES IN THE SPENT FUEL POOL

### Summary:

During Turkey Point refueling outages, wet annular burnable absorber (WABA) assemblies are discharged in the spent fuel pool and placed in to spent fuel assemblies. This evaluation documents the use of funnel device to ease the installation of spent WABAs into spent fuel assemblies in the Turkey Point Units 3 and 4 spent fuel pools. The proposed use of the WABA lead in or funnel devices involves the following changes to the plant for permanently discharged fuel: 1) a change in the fuel assembly weight and 2) a change in the interface of the WABA assembly with the fuel assembly. Evaluation of effects on safety were discussed in detail for criticality, corrosion, flow blockage and cooling, structural integrity of the funnel device, structural impact on the Spent Fuel Pool, and handling of the fuel assembly with the WABA insert and funnel device.

### Safety Evaluation:

This evaluation concluded that that the use of WABA funnel devices in the spent fuel pool does not adversely affect pool subcriticality, fuel assembly cooling, pool structural integrity, or the potential for a fuel handling accident. The evaluation further concluded that the use of WABA funnel devices as instructed will not impact plant safety or plant operations, and therefore, does not constitute an unreviewed safety question or require changes to the plant technical specifications. Thus, prior NRC approval was not required for installation of the WABA funnel devices identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SECS-99-019**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 06/08/99 |

**FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 5 EXCLUDING APPENDIX 5E**

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC, L-97-143 dated July 1, 1997.

FSAR Chapter 5 provides an overview of the plant buildings including safety related structures. The review of Chapter 5 identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review. The evaluation to address any proposed revisions to Appendix 5E of the FSAR has been performed separately.

**Safety Evaluation:**

This review has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not constitute an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION PTN-ENG-SECS-99-020**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 06/20/99 |

## FSAR ACCURACY REVIEW CHANGES FOR APPENDIX 5E

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143, dated July 1, 1997.

Appendix 5E of the FSAR provides a description of the design basis missiles for Turkey Point Units 3 and 4 and outlines the protective measures taken against unacceptable damage due to missile impact for vital structures and components including safety related structures. The review of Appendix 5E identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

**Safety Evaluation:**

This review has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not constitute an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

## SAFETY EVALUATION PTN-ENG-SENS-99-024
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 12/30/99

## LEAK INSPECTION OF RHR (PIGGY-BACK) RECIRCULATION FLOW PATHS

### Summary:

This safety evaluation addressed the impact on plant safety and operation associated with performing a leak test of the post-LOCA "piggy-back" recirculation flow path at near residual heat removal (RHR) pump shutoff head conditions. The testing was required to satisfy FSAR commitments related to relative leak tightness of the external recirculation loop piping. The test boundary addressed in this safety evaluation extended from valves MOV-*-863A/B to the suction of the high head safety injection (HHSI) pumps and containment spray pumps. The evaluation established the appropriate test pressure, RHR pump operating requirements, system alignments, and plant restrictions that must be met to perform the test. It also addressed the ability to conduct the test with an external pressure source in lieu of using the discharge head of an operating RHR pumps. Since any recirculation loop leakage in the test mode would represent a reduction in reactor coolant system inventory, this safety evaluation required that the unit in test be defueled prior to performing the test with an RHR pump.

An FSAR change package was provided as an Attachment to this evaluation to allow the use of an external pressure source, in lieu of the installed RHR pumps, to pressurize portions of the recirculation piping.

Revision 1 identified the procedures which currently exist to satisfy the technical specification requirements and testing at near pump shutoff head conditions. It also identified those procedures that still need to be developed to comply with the FSAR requirement when an external pressure source is used.

### Safety Evaluation:

This evaluation defined the requirements needed to satisfy the FSAR leak inspection commitment. An assessment of the fluid conditions concluded that the test would not adversely impact the integrity of any component included within the test boundary. The actions or temporary plant conditions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or temporary plant conditions identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SECS-99-025
REVISION 2

UNIT : 4
APPROVAL DATE : 04/15/98

## STORAGE OF TOOLS AND EQUIPMENT IN CONTAINMENT DURING ALL MODES OF OPERATION

### Summary:

This evaluation addressed the acceptability of leaving a quantity of tools and equipment within the Unit 4 containment structure during all modes of plant operation. The items to be stored, and the storage locations within the Unit 4 containment, were specifically identified within the evaluation. The purpose of leaving these tools and equipment within containment following refueling outages was to reduce the usage demand on the Unit 4 polar crane during refueling outages. This evaluation considered the potential for adverse seismic interactions with safety related equipment, the potential for additional hydrogen generation within containment during accidents, the impact on the containment free volume and heat sink analyses, the potential to obstruct flow to the containment sumps, and the impact on containment combustible loading. To ensure that the tools and equipment addressed in the evaluation were safely stored during plant operation, both generic and specific actions and restrictions were identified for implementation within the evaluation.

Revision 2 evaluated the impact of leaving two additional items in containment. These additional items were 100 hp electric motors left inside containment following maintenance on the 4A normal containment cooler.

### Safety Evaluation:

The safety evaluation concluded that the proposed items identified within the safety evaluation can safely remain within containment during all modes of operation, provided that all the restrictions and requirements identified within the evaluation were implemented following each outage. The evaluation further concluded that the identified restrictions and requirements would ensure that these activities would have no adverse effects on plant operation, and would not compromise the safety and licensing bases for Unit 4. Consequently, the requirements and restrictions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the requirements or restrictions identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SENS-99-050

REVISION 0

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| UNITS | : | 3 & 4 |
| APPROVAL DATE | : | 06/28/99 |

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 11 AND SECTION 14.1.13

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143, dated July 1, 1997.

Chapter 11 provides a description of the waste disposal and radiation protection system. Section 14.1.13 discusses the turbine control system. The review of Chapter 11 and Section 14.1.13 identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

The activity being performed by this evaluation updates the documentation in the FSAR to accurately reflect the facility in terms of how it is operated and what equipment is present in the field. No physical changes are being made to the facility or its manner of operation. These documentation changes have been evaluated and do not identify cases where the field condition is inappropriate or in conflict with the plants design bases.

**Safety Evaluation:**

The FSAR changes proposed in this evaluation have been evaluated under 10 CFR 50.59 and found not to require any changes to the technical specifications and not to involve an unreviewed safety question. Accordingly, these changes may be implemented without prior NRC approval.

## SAFETY EVALUATION PTN-ENG-SENS-99-059
REVISION 0

UNIT : 4
APPROVAL DATE : 05/13/99

## SAFETY EVALUATION FOR REMOVAL OF PRESSURIZER HEATERS FROM SERVICE

**Summary:**

This evaluation addressed continued plant operation with a pressurizer heater out of service. It was determined that sufficient pressurizer heater capacity existed with heater No. 63 out of service to meet the technical specification requirements for pressurizer heater capacity, and the operational requirement for total pressurizer heater capacity specified in the plant operating procedures. The evaluation was based on a Westinghouse analyses of operating transients in support of the Turkey Point Units 3 & 4 power uprate effort. The referenced analysis utilized a pressurizer heater capacity of 1000 kW and then concluded that the results would not be affected by a 50% reduction in heater capacity. The as-built capacity of the Turkey Point pressurizer heaters was 1300 kW.

An FSAR change package was provided as an attachment to this safety evaluation. The FSAR was revised to clarify the distinction between the original design capacity and the analyzed capacity reflected in the thermal power uprate analyses.

**Safety Evaluation:**

This evaluation determined that operation of the pressurizer with heaters out of service is acceptable and the specific removal of one heater posed no adverse affect on plant safety. The evaluation concluded that the proposed plant configuration and mode of operation was bounded by the technical specifications and did not change the analysis of accidents addressed in the FSAR or the results and conclusions of any previous safety evaluations. The actions or precautions identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions and documentation changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or precautions identified in this safety evaluation.

# SAFETY EVALUATION PTN-ENG-SEIS-99-062
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 07/28/99

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 7 [SECTIONS 7.2 AND 7.5]

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143, dated July 1, 1997.

FSAR Chapter 7, Sections 7.2 and 7.5, involve the reactor protection system and engineered safety features actuation system. The review of these Chapter 7 sections identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

**Safety Evaluation:**

This review has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not constitute an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION PTN-ENG-SENS-99-066**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 12/16/99 |

## SAFETY EVALUATION FOR SECONDARY BARRIER CONTAINMENT INTEGRITY FUNCTION FOR PENETRATIONS 27A, 27B, AND 27C


**Summary:**

This evaluation was developed to determine if proposed changes in containment isolation assignments applied to the Turkey Point Units 3 and 4 Train 1 auxiliary feedwater (AFW) discharge check valves (*-20-140, *-20-240 and *-20-340) to the associated upstream AFW flow control valves (CV-*-2816, CV-*-2817, and CV-*-2818, respectively) represented an unreviewed safety question. The proposed changes were specifically evaluated with respect to the requirements of the 1967 proposed General Design Criterion 53, 10 CFR 50 Appendix J - *Primary Reactor Containment Leakage Testing*, ASME Section XI Inservice Testing, and Turkey Point Units 3 & 4 AFW system design requirements.

Design Basis Document (DBD) and FSAR change packages were provided as attachments to this safety evaluation to document the revised containment isolation barrier assignments.


**Safety Evaluation:**

This safety evaluation addressed the technical and licensing aspects of re-assigning the outside containment isolation barrier function for penetrations 27A, 27B and 27C, and provided for updating the FSAR and DBD accordingly. The proposed change did not modify the design or configuration of the penetration and did not introduce any new hazards. It concluded that the FSAR commitment that containment isolation be established assuming an independent single active failure remained intact for the penetration, and that the applicable documentation changes did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the containment isolation barrier changes.

## SAFETY EVALUATION PTN-ENG-SENS-99-072
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 12/16/99 |

## SAFETY EVALUATION FOR USE OF MANUAL ACTIONS TO RE-ALIGN 3D/4D BUS FOR RHR LOOP OPERABILITY IN MODE 6

### Summary:

This safety evaluation examined the potential impact on plant safety and operation associated with a minor procedure change in the definition of an operable RHR loop when the plant is in Mode 6 with 23 feet of water above the reactor vessel flange. The current definition required that the RHR pump, CCW pump, and ICW pump of an operable cooling loop be powered from the same electrical train. The proposed definition allowed credit to be taken for the swing capability of the "C" ICW pump and "C" CCW pump given that the 3D/4D bus can be powered from either the "A" or "B" electrical train. That is, action would not be required to pre-align the "C" ICW pump and "C" CCW pump to the associated RHR train if they were being used to support RHR loop operability. The 3D/4D bus is normally aligned to the "B" electrical train. Alignment to the "A" electrical train requires manual operator action outside the control room.

Relaxation of the current procedural restriction for Mode 6 with high water in the refueling cavity required consideration of the time it takes to perform the manual actions, and the thermal margin available in Mode 6 with 23 feet of water above the reactor vessel flange.

### Safety Evaluation:

This evaluation demonstrated that the use of proceduralized manual actions to re-align the 3D/4D bus to support RHR loop operability did not cause any safety limits or design limits to be exceeded. It also demonstrated that there are no postulated events or conditions that would prevent the manual actions from being accomplished when required. Since the proposed actions satisfied all design and licensing requirements, it was concluded that the change in operating strategy did not constitute an unreviewed safety question or require a change to the technical specifications. Thus, NRC approval was not required for implementation of associated procedural change.

**SAFETY EVALUATION PTN-ENG-SEIS-99-099**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 8/26/99 |

**FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 7**
**[SECTIONS 7.3, 7.4 AND 7.6]**

**Summary:**

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143, dated July 1, 1997.

Chapter 7, Sections 7.3, 7.4, and 7.6 describe regulating systems, nuclear instrumentation, and in-core instrumentation. The review of Chapter 7, Sections 7.3, 7.4, and 7.6 identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

The activity being performed by this evaluation updates the documentation in the FSAR to accurately reflect the facility in terms of how it is operated and what equipment is present in the field. No physical changes are being made to the facility or its manner of operation. These documentation changes have been evaluated and do not identify cases where the field condition is inappropriate or in conflict with the plants design bases.

**Safety Evaluation:**

This evaluation has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not involve an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

**SAFETY EVALUATION JPN-PTN-SEIS-99-102**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 08/26/99 |

## FSAR ACCURACY REVIEW CHANGES FOR CHAPTER 7 [APPENDIX 7A]

### Summary:

Industry events and regulatory concerns have resulted in an increased emphasis by the NRC on the accuracy of the facility and procedure descriptions in the Final Safety Analysis Reports (FSAR). FPL has performed several self-assessments of the Turkey Point FSAR for accuracy over the last several years. Although these self-assessments did not identify significant concerns, a number of FSAR discrepancies were identified. In the Turkey Point response to the NRC's 10 CFR 50.54(f) request for information regarding the adequacy of and availability of design basis information, FPL committed to perform an FSAR assessment using an approach outlined in NEI 96-05. FPL also committed to perform an additional detailed review of portions of the FSAR over a two year period to identify and correct documentation discrepancies. The scope of the detailed review of the entire FSAR is described in FPL letter to the NRC L-97-143, dated July 1, 1997.

Chapter 7, Appendix 7A describes the safety assessment system. The review of Chapter 7, Appendix 7A identified a number of editorial discrepancies and minor technical discrepancies. No operability issues were identified as a result of this review.

The activity being performed by this evaluation updates the documentation in the FSAR to accurately reflect the facility in terms of how it is operated and what equipment is present in the field. No physical changes are being made to the facility or its manner of operation. These documentation changes have been evaluated and do not identify cases where the field condition is inappropriate or in conflict with the plants design bases.

### Safety Evaluation:

This evaluation has determined that the identified FSAR discrepancies do not impact safe operation of the plant, do not involve an unreviewed safety question and do not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification can be made and do not require NRC approval prior to implementation.

## SAFETY EVALUATION PTN-ENG-SEYS-99-106
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 05/24/00 |

## SAFETY EVALUATION FOR NON-VITAL BATTERY TEST


### Summary:

This safety evaluation was written to support a Temporary Procedure (TP 99-028) to conduct an IEEE Standard 450 Performance Test on the Non-Vital Station Battery 4D34. The purpose of the test was to verify that the battery (which provides emergency power to DC Bus 4D31) would perform if challenged during a Loss of Off-site Power (LOOP) event. Station battery 4D34 was in its 17th year of a designed 20-year useful life and no data existed to confirm its capacity. The TP was written to provide a partial discharge of the 4D34 battery without diminishing the operability of the 4D31 bus. Due to the similarity between the 4D31 bus and the 3D31 bus, the safety evaluation was applicable to a performance test on the 3D34 battery.

The 3D31 and 4D31 dc buses provide power to non-safety plant switchgear, load centers, non-safety inverters, and non-safety emergency turbine lube oil pumps. A possible loss of the 4B steam generator feedwater pump (SGFP) and its potential failure to trip during the discharge test was addressed in the evaluation since breaker tip contacts for the SGFP provide an input signal to the auxiliary feedwater system auto start circuitry.


### Safety Evaluation:

This evaluation examined the minimum voltage requirements of each load on the 4D31 bus, the condition of the battery at the conclusion of the test, the impact of testing on plant operation, failure modes and effects, plant operating restrictions during testing, and applicable compensatory measures. The evaluation concluded that the proposed testing approach and test equipment had no adverse impact on plant safety or plant operations, and therefore, did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to perform the subject temporary procedure.

## SAFETY EVALUATION PTN-ENG-SEMS-99-117
REVISION 0

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| UNITS | : | 3 & 4 |
| APPROVAL DATE | : | 02/23/00 |

## SAFETY EVALUATION FOR FIRE PROTECTION SELF-ASSESSMENT DOCUMENT CHANGE

**Summary:**

During a fire protection self-assessment completed on September 30, 1999, discrepancies were noted which in some cases would require FSAR or other document changes for resolution. Opportunities were found to augment and clarify design basis documentation, to correct or enhance consistency among design documents and to streamline and clarify procedural requirements. Fire protection and 10 CFR 50 Appendix R Safe Shutdown Features are classified as Quality Related. There was no impact on maintaining reactor coolant system pressure boundary integrity, achieving and maintaining safe shutdown, or the capability to mitigate accidents with radioactivity releases approaching 10 CFR 100 limits. The purpose of this safety evaluation was to provide appropriate bases and reviews pursuant to 10 CFR 50.59 to support UFSAR and other document changes associated with fire protection and safe shutdown capability.

The required FSAR changes were documented in an attachment to the evaluation.

**Safety Evaluation:**

All identified discrepancies were dispositioned and none were found to impact nuclear safety or safe plant operation. Updates were prepared to correct and clarify the FSAR. Engineering assessment of the findings determined that no operability issues were involved. The actions and FSAR changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the actions or changes identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-00-001
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 02/03/00

## SAFETY EVALUATION FOR USE OF ALTERNATE AMINES

### Summary:

This evaluation provided the technical justification to use alternate amines in lieu of ammonia for secondary system fluid chemistry pH and erosion-corrosion control. It addressed the physical location and configuration of alternate amine chemical facilities, the chemical injection process and the expected effects on the plant secondary system piping and components. The chemicals considered were Ethanalomine (ETA) and Methoxypropylamine (MPA) in concentrations from 0 to 10 ppm. Use of a less volatile amine such as ETA or MPA will better distribute a basic pH throughout the secondary system and is expected to reduce iron transport rates, which in turn should reduce sludge deposition and piping corrosion rates in the secondary system.

An FSAR change package was provided as an attachment to this evaluation. The change package revised the FSAR text to allow ammonium hydroxide or an alternate amine to be used to control secondary system pH.

### Safety Evaluation:

This safety evaluation demonstrated that the storage, processing and delivery of an alternate amine solution to the secondary system is similar to that of existing secondary system water treatment chemicals. No new hazards are created by the alternate treatment scheme and there is no reduction in system piping or component reliability. The actions and FSAR changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the actions or changes identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SEFJ-00-007**
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 05/24/00

**FSAR AND DBD CHANGES CORRESPONDING TO THE LOSS OF LOAD REANALYSIS FOR UPDATES IN SURGE LINE RESISTANCE AND PRESSURIZER LEVEL UNCERTAINTY**


**Summary:**

The Loss of Load design basis analysis was reanalyzed to include uncertainty for the pressurizer initial water level. Previous to this, the Loss of Load design basis event had been analyzed with nominal pressurizer water level with no uncertainties included. This was not consistent with the Westinghouse Safety Analysis standards recommendation that the maximum water level (with uncertainties added in the conservative direction) be used in the analysis. In addition, updates to the pressurizer surge line hydraulic resistance were included. The results of the Loss of Load reanalysis showed that all of the applicable safety acceptance criteria continued to be met.

Revision 1 incorporated the proposed changes to the accident analysis DBD. The recommended FSAR changes were included in Attachment 1. The recommended DBD changes were included in Attachment 2.


**Safety Evaluation:**

The proposed FSAR and DBD changes summarized the methodology and the results of the revised Loss of Load safety analysis. The document updates did not alter the sequence of events during the accident or the equipment required to mitigate its consequences. Since the appropriate safety analysis acceptance criteria continue to be met, the document changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-00-007
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 03/03/00

## SAFETY EVALUATION FOR SFP TEMPERATURES DURING OFFLOAD STARTING BEFORE 150 HOURS


**Summary:**

This evaluation provided for document changes associated with spent fuel pool (SFP) temperature and refueling administrative controls. The purpose of this evaluation was to determine if core offloads for Turkey Point Units 3 and 4 could commence sooner than 150 hours after shutdown as was required by administrative procedures. This evaluation examined the impact of various start times and fuel transfer rates on SFP heatup. It also addressed issues such as SFP temperature overshoot potential, SFP pre-heating due to higher CCW temperatures during the plant cooldown, and the impact of higher fuel assembly decay heat rates on fuel clad temperature.

A new offload start time of 130 hours was selected based on the analysis. This change provided added flexibility in selecting times for starting and completing fuel transfer to the spent fuel pool during refueling outages.

An FSAR change package was provided as an attachment to the evaluation.


**Safety Evaluation:**

The safety evaluation analyzed shorter refueling outage start times with respect to the existing FSAR analysis. It also provided the necessary documentation to update the FSAR description of a typical refueling sequence. The safety evaluation did not implement any changes to the plant configuration, method of handling and storing irradiated fuel assemblies, spent fuel pool accessibility, or design bases for facilities. Since the proposed actions and documentation changes did not impact any licensing commitments associated with spent fuel pool operation, the safety evaluation did not constitute an unreviewed safety question or require a change to the plant technical specifications Therefore, implementation of the proposed actions and documentation changes did not require prior NRC approval.

**SAFETY EVALUATION PTN-ENG-SEMS-00-008**
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 02/04/00

**SAFETY EVALUATION FOR**
**USE OF PRC-01 RESIN TO REMOVE COBALT-58**
**CONTAMINANTS IN THE LETDOWN STREAM**

**Summary:**

This evaluation assessed the acceptability of applying PRC-01 resin in the chemical and volume control system (CVCS) demineralizers for the purpose of limiting cobalt-58 particles released during refueling outage crud burst activities. Analyses of crud burst activities during recent outages identified cobalt-58 particles as a major contributor to heightened personnel dose rates. Cobalt-58 particles sized on the order of 0.05 - 0.1 microns could not be removed by standard CVCS letdown mixed bed demineralizers and filters following crud bursts. This evaluation selected the PRC-01 resin as the best method for enhancing CVCS cleanup activities following a crud burst. This evaluation did not justify use of the resin during any operating condition however, the resin may be loaded while the CVCS is in normal operation.

Attachment 1 to this evaluation provides the radiation stability data and composition of the PRC-01 media. Attachment 2 assesses the maximum contaminant concentrations anticipated as a result of resin release and breakdown in the reactor coolant system (RCS) during crud burst activities. Attachment 3 provides the vendor response to various technical issues associated with use of PRC-01.

**Safety Evaluation:**

The safety evaluation concluded that the application of PRC-01 resin media to cleanup the letdown stream during crud burst activities was acceptable provided that all of the restrictions and requirements identified within the evaluation were implemented. The evaluation further concluded that the identified restrictions and requirements would ensure that the use of PRC-01 would have no adverse effects on plant operation, and would not compromise the safety and licensing bases for Units 3 and 4. Consequently, the requirements and restrictions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the requirements or restrictions identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SEFJ-00-012**
REVISION 0

UNIT : 4
APPROVAL DATE : 08/31/00

**TEMPERATURE / POWER COASTDOWN FOR TURKEY POINT UNIT 4 CYCLE 18**

## Summary:

The purpose of this evaluation is to support a temperature/power coastdown at the end of Cycle 18 for Turkey Point Unit 4. The proposed coastdown is needed to overcome a cycle energy shortfall and allow the plant to continue operation until the scheduled end-of-cycle date of September 25, 2000. The proposed coastdown consists of a 5°F reduction in RCS Tavg at a rate of approximately 1°F per day followed by a power reduction of about 5% at a rate of approximately 1.3% a day. The total cycle exposure is not to exceed 12,832 effective full power hours.

The purpose of this safety evaluation was to allow Turkey Point Unit 4 to extend the length of Cycle 18, after reaching the end of reactivity at nominal hot full power conditions, by using a combined $T_{avg}$ and power coastdown. The reactivity necessary to extend operation was obtained from the negative moderator temperature and power coefficients. Because these coefficients were negative, a decrease in either the moderator average temperature or the reactor power would result in a positive reactivity addition to the core, offsetting the reactivity loss from burnup. The $T_{avg}$/power coastdown was started at the end of Cycle 18 by reducing primary $T_{avg}$ by 5°F at a rate of approximately 1°F per day. The temperature coastdown was followed by a power reduction of 5% at a rate of approximately 1.3% per day.

## Safety Evaluation:

This safety evaluation demonstrated that the planned temperature/power coastdown did not affect any assumptions relative to accident initiators, did not impede the accomplishment of post-accident recovery efforts, or increase the consequences of postulated accidents. Since plant design requirements continued to be met and the integrity of the reactor coolant system pressure boundary was not challenged, it was concluded that the assumptions employed in the calculation of offsite radiological doses remained valid. The actions or plant procedure changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to initiate the temperature/power coastdown.

## SAFETY EVALUATION PTN-ENG-SENS-00-013

REVISION 0

UNIT : 4
APPROVAL DATE : 02/18/00

## SAFETY EVALUATION FOR PLANT OPERATION WITHOUT THE STEAM GENERATOR FEEDWATER PUMP TRIP INPUT SIGNAL TO THE AFW ACTUATION LOGIC

### Summary:

This safety evaluation was written to support a Temporary Procedure (TP 00-009) to de-energize the 4D31 non-vital dc bus for maintenance. De-energizing the 4D31 bus would remove control power from the 4B steam generator feedwater pump breaker, which in turn, would prevent the breaker from opening under all trip conditions. Since an auxiliary contact on the feedwater pump breaker is used to actuate the auxiliary feedwater (AFW) system, this evaluation used the criteria of 10 CFR 50.59 to determine if temporarily removing the steam generator feedwater pump trip from the AFW system actuation logic represented an unreviewed safety question, or required a change to the plant technical specifications. The AFW system provides a post-accident heat removal function during certain design basis events.

### Safety Evaluation:

This evaluation reviewed each of the accident scenarios that rely on AFW flow delivery and determined that AFW actuation from the feed pump breakers is not credited in any of the supporting analyses. The review also demonstrated that the feed water control valves would be able to affect feedwater isolation in the applicable scenarios if the pump trip breaker failed to open when required. It was concluded that the actions and temporary plant conditions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or temporary plant conditions identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEFJ-00-015
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 03/02/99 |

## TECHNICAL SPECIFICATION AND FSAR CHANGES ASSOCIATED WITH SOLUBLE BORON CREDIT FOR SPENT FUEL STORAGE AND NEW FUEL CRITICALITY ANALYSIS


### Summary:

This safety evaluation was prepared to update the Turkey Point FSAR and Technical Specification (TS) Bases Document to reflect NRC approved changes to fuel storage criticality that: a) allow the use of soluble boron credit for the storage of fresh and irradiated fuel in the spent fuel pool (SFP), and b) make the analytical method used for the storage of fuel in the new fuel storage room consistent with that used in the SFP. Additionally, a change to the FSAR regarding the effect of no Boraflex on the periphery of Region II storage modules was evaluated to determine its impact on safe plant operation.

Attachment 1 to this evaluation provided the proposed changes to the TS Bases Document. Attachment 2 provided the recommended FSAR changes.


### Safety Evaluation:

The proposed FSAR and TS Bases changes summarized the methodology and results of fuel storage criticality analyses that were reviewed and approved by the NRC. The document updates did not alter the sequence of events of postulated criticality accidents, or the equipment required to mitigate associated radiological consequences. Consequently, the document changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SENS-00-015**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 02/24/00 |

## SAFETY EVALUATION FOR RESIDUAL HEAT REMOVAL SAMPLING IN MODES 1, 2, AND 3

### Summary:

Plant procedures require that fluid samples of the residual heat removal (RHR) system be taken prior to placing the system in service during a unit shutdown. The fluid samples are analyzed to ensure that the water chemistry is compatible with the reactor coolant system boron and lithium concentrations prior to initiating shutdown cooling, and to verify that the integrity of the RHR heat exchanger tubes have not been breached since the previous unit shutdown. The current sampling procedure is performed when the plant reaches Mode 4 during the shutdown process. The sampling procedure requires that a manual RHR isolation valve be temporarily opened.

Sampling the RHR system late in the shut down process has the potential to delay a cooldown to cold shutdown conditions, especially if chemistry changes or the repair of a RHR heat exchanger is necessary to restore system operability. In an effort to avoid potential shutdown delays, this safety evaluation examines the use of administrative controls to permit timely isolation of the RHR pressure boundary during an accident such that the sampling procedure can be performed while the plant is operating in Modes 1, 2, or 3. The criteria of 10 CFR 50.59 is used to demonstrate that the revised sampling procedure and administrative controls do not pose an unreviewed safety question or requires changes to the plant technical specifications.

### Safety Evaluation:

This safety evaluation addressed the impact on plant safety associated with temporarily opening a RHR pressure boundary isolation valve while the system is aligned in standby for low head safety injection service. It specifically analyzed the affects on safety associated with manual restoration of the RHR system during accident conditions and demonstrated that the RHR system would be maintained in an analyzed configuration during an event; capable of accomplishing its flow delivery function. The actions and precautions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the identified actions and precautions.

**SAFETY EVALUATION PTN-ENG-SEFJ-00-019**
REVISION 0

UNIT : 4
APPROVAL DATE : 09/07/00

## SAFETY EVALUATION FOR FUEL ASSEMBLY TOP NOZZLE EXAMINATION AND REPLACEMENT IN THE UNIT 4 SPENT FUEL POOL


### Summary:

In December 1999, Westinghouse issued WCAP-15356, "Top Nozzle Holdown Spring Screw Fracture, Root Cause Final Report," providing the root cause for top nozzle spring fractures of fuel assemblies. Based on Westinghouse recommendations, Unit 4 was to perform spring scale testing and/or visual inspection of all twice burnt fuel assemblies that were to be used in the next cycle. This evaluation provided justification for the temporary relocation of irradiated fuel assemblies and the use of fuel assembly inspection and fuel assembly top nozzle replacement equipment in the Unit 4 spent fuel pool (SFP). Underwater cameras and spring pulling tools were to be used to perform fuel assembly spring scale test and / or visual examinations of fuel assemblies. Based on examination results, fuel assembly top nozzles would be replaced, as needed, prior to being returned to the reactor core. Top nozzle replacement would be performed in the storage rack of the SFP using tools on extension poles and rigging to aid in the operation.


### Safety Evaluation:

This evaluation concluded that the proposed fuel assembly and top nozzle examination replacement activities could be performed in the Turkey Point Unit 4 Spent Fuel Pool. It was determined that the proposed activities would not constitute an unreviewed safety question and would not require any changes to the plant technical specifications. Furthermore, the proposed activities would not adversely affect plant safety, fuel integrity, the performance of refueling activities, or the safe operation of the SFP. Thus, prior NRC approval was not required to perform the inspection-related activities in the SFP.

**SAFETY EVALUATION PTN-ENG-SEFJ-00-020**
REVISION 0

UNIT : 3
APPROVAL DATE : 10/31/00

# SAFETY EVALUATION FOR UNIT 3 SPENT FUEL POOL BORAFLEX SURVEILLANCE USING THE BADGER METHOD


## Summary:

This safety evaluation was developed to establish the technical justification to support blackness testing of the Unit 3 spent fuel storage racks. Blackness testing is a technique used to measure the level of neutron absorption (degree of blackness) of the spent fuel racks with Boraflex or other neutron absorbing material(s) installed. The design of the Unit 3 racks includes the use of Boraflex material, which is a strong neutron absorption material used to maintain the spent fuel in the pool in a subcritical condition. Boraflex degradation has been an issue since 1987, and extensive industry study of the degradation of the Boraflex material has utilized the techniques associated with blackness testing and analysis of coupons. This safety evaluation addresses blackness testing using the BADGER (**B**oron-10 **A**erial **D**ensity **G**age for **E**valuating **R**acks) technique and the interaction of the test equipment with the spent fuel and spent fuel storage racks.


## Safety Evaluation:

This evaluation examined the proposed blackness testing technique specified in the evaluation and determined that it would not violate technical specification requirements for the spent fuel pool, would not violate heavy load requirements, and would not alter any margin of safety associated with the prevention and mitigation of fuel handling accidents. Consequently, the proposed testing preparations and implementation requirements identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the testing requirements identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SENS-00-021**
REVISION 0

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| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 03/16/00 |

## SAFETY EVALUATION FOR INSTALLATION OF FRAMATOME COGEMA FUELS ROD CLUSTER CONTROL ASSEMBLIES (RCCAs)

**Summary:**

The purpose of this safety evaluation was to document the design requirements for the installation of new Framatome Cogema Fuels 15 x 15 rod cluster control assemblies (RCCAs) in Turkey Point Units 3 and 4 and to identify consequential changes to the FSAR. Since some of the design features of the new RCCAs were different from the current RCCAs in-service, it was necessary to document compliance, providing evidence that each of the design requirements were met prior to fuel installation.

The recommended FSAR changes were provided as Attachment 1 to the evaluation.

**Safety Evaluation:**

The results of this Safety Evaluation concluded that the new RCCA design met the requirements of Framatome and that the proposed change would not have an adverse effect on plant safety or operation, would not constitute an unreviewed safety question, and would not require changes to the technical specifications. Therefore, prior NRC approval was not required.

## SAFETY EVALUATION PTN-ENG-SEMS-00-023

REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/31/00

## SAFETY EVALUATION FOR SERVICE AIR PENETRATION 34 DESIGN BASIS SAFETY CLASSIFICATIONS

### Summary:

This safety evaluation provides the supporting documentation necessary to exempt check valve *-40-205 from functional testing every time it is opened and inspected or cleaned. Check valve *-40-205 provides the containment isolation barrier inside containment for penetration No. 34. This penetration is for the process line providing service air into containment for maintenance-related activities. It also provides the flow path for post-LOCA hydrogen removal and containment dilution for hydrogen control. Existing documentation required that the check valve be subjected to local leak rate testing for its containment isolation function, functional testing to open for its hydrogen recombination function and post-maintenance inspection after valve disassembly. For many years, local leak rate testing of the subject check valves has been hampered because particulate matter, transported by the service air, prevents proper disc seating. Such particulate matter transport can occur during air supply service inside containment or during check valve functional testing.

This evaluation reviewed the safety classifications of the various penetration flow path functions and established appropriate test requirements to comply with regulatory and licensing commitments. The recommended FSAR, Design Basis Document, and procedural changes were included in the evaluation.

### Safety Evaluation:

This safety evaluation examined the technical and licensing aspects associated with reducing the post maintenance test requirements for valve *-40-205, and provided for updating the FSAR and DBD accordingly. The proposed change did not modify the design or configuration of penetration No. 34 and did not introduce any new hazards. It concluded that all applicable regulatory and licensing commitments would continue to be met for the penetration, and that the applicable documentation changes did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the testing changes.

**SAFETY EVALUATION SENS-00-046**
REVISION 1

| | | |
|---|---|---|
| UNIT | : | 4 |
| APPROVAL DATE | : | 08/08/00 |

## SAFETY EVALUATION FOR TEMPORARY LOWERING OF UNIT 4 SPENT FUEL POOL WATER LEVEL FOR MAINTENANCE ACTIVITIES


**Summary:**

This evaluation was developed to examine the effects of securing the spent fuel cooling pumps and reducing the pool level by about 1-foot in order to perform maintenance on valve 4-821 in the primary water system makeup line to the spent fuel pool (SFP). This evaluation addressed the effects of spent fuel handling accidents, spent fuel heatup rates; increased radiation levels resulting from lowered water (shielding) levels, and activation of system alarms. To reduce the potential for fuel handling accidents, all fuel movement and crane operation was suspended in accordance with Technical Specification 3/4.9.11. The spent fuel pool has been evaluated for elevated pool temperatures and pool heatup from 100 °F to 135 °F was estimated to take about 18 hours, which would be a sufficient time to perform the required maintenance. Previous evaluations of reduced water levels have demonstrated that expected increases in radiation levels would be negligible. In order to preclude activation of the SFP alarms, pool temperature and level were required to be monitored on an hourly basis. A SFP temperature limit of 130 °F was established as an upper limit during the maintenance activity, at which time work would be secured and SFP cooling restored.


**Safety Evaluation:**

This evaluation concluded that reducing the spent fuel pool level for maintenance on the primary water makeup valve would not adversely impact plant operation and would not compromise the spent fuel handling accident analyses, provided that the actions and restrictions identified in the evaluation were observed. Consequently, the reduced pool water level and other actions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-00-050
REVISION 0

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 09/19/00 |

## SAFETY EVALUATION FOR ELIMINATION OF TURBINE BUILDING EL. 18 FT. FIRE WATCH PATROL

**Summary:**

This safety evaluation provided the necessary technical and licensing justification to delete the hourly fire watch patrol on the 18 foot elevation of the turbine building, and to update the FSAR accordingly. The fire protection program has required an hourly fire watch patrol of the El. 18' of the turbine building since 1979. At the time, the NRC requested FPL to extend the turbine building ground floor sprinklers to provide exposure protection for cable trays where safe shutdown circuits were located. In lieu of sprinkler modifications, FPL proposed to enhance control of transient combustibles and institute the fire watch patrol. The NRC accepted this approach and issued the fire protection SER in 1979, requiring implementation of the fire watch patrol. Since the basis for the original concern with the El. 18' of the turbine building was protection of safe shutdown capability and since Turkey Point has assured this capability through the implementation of 10 CFR 50 Appendix R commitments, the basis for the concern no longer exists.

This evaluation does not apply to fire watches established as a compensatory measure due to degraded Thermo-Lag fire barriers.

The recommended FSAR changes were provided as an attachment to the evaluation.

**Safety Evaluation:**

The requirement for the fire watch came from historical concerns, which have since been addressed. Existing fire barriers or spatial separation protect safe shutdown components or circuits in combination with the open nature of the structure, which would prevent the buildup of products of combustion. These features coupled with the existence of fire fighting equipment and the trained site fire brigade, assure that safe shutdown capability is protected. As such there is no decrease in the effectiveness of the fire protection program. The proposed documentation changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, do not constitute an unreviewed safety question, do not require a change to the plant technical specifications and do not change the fire hazards analysis basis for postulated fires in the affected fire zones. Therefore, the proposed changes do not require prior NRC approval.

**SAFETY EVALUATION PTN-ENG-SENS-00-058**
REVISION 1

UNIT : 4
APPROVAL DATE : 09/01/00

## SAFETY EVALUATION FOR ON-LINE REPLACEMENT OF THE 4B RHR PUMP SHAFT SEAL


**Summary:**

This safety evaluation analyzed the maintenance activities associated with replacing a leaking residual heat removal (RHR) system pump shaft seal while the plant is operating in Mode 1. The RHR system supports the low head safety injection function during plant operating Modes 1, 2, and 3, and is classified as safety related. The maintenance evolution involved removal of the affected RHR pump from service and temporary modification of the auxiliary building shielding. Appropriate precautions and limitations were established to: a) ensure that the redundant RHR train remained operable and fully qualified during the maintenance evolution, and b) ensure that any post-accident recovery actions required in and around the auxiliary building would not be hampered by elevated post-accident dose rates.


**Safety Evaluation:**

This evaluation determined that replacement of a defective RHR pump seal on-line was a prudent and safety focused maintenance activity. It concluded that the activity would not adversely impact plant operation or post-accident recovery efforts, provided that the actions and restrictions identified in the evaluation were observed. Consequently, the actions and restrictions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to perform the seal replacement activity.

**SAFETY EVALUATION PTN-ENG-SEMS-00-067**
REVISION 0

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 08/31/00 |

## SAFETY EVALUATION FOR CVCS HOLDUP TANKS VACUUM BREAKERS

**Summary:**

This safety evaluation was prepared to resolve an apparent discrepancy between the cracking pressure of the vacuum breakers installed on the chemical and volume control system (CVCS) holdup tanks (HUTs) and the cracking pressure value specified in Table 9.2-3 of the FSAR. Per vendor data, the installed vacuum breakers have a cracking pressure (pressure at which the valve starts to open) of 4.3 in.-water column (WC) instead of 3.0 in.-WC as specified in the FSAR. The evaluation determined that the reference cracking pressure of 4.3 in.-WC is a nominal value and that the actual cracking pressure of the vacuum breakers is much lower. Calculations performed by the valve vendor indicated that the actual cracking pressure of a standard 4-inch vacuum breaker mounted horizontally in a configuration similar to that on the HUTs would be 1.6 in.-WC ±10%. Since the external design pressure for the HUTs is 4.3 in.-WC, the installed vacuum breakers were determined to be acceptable. The evaluation used the criteria of 10 CFR 50.59 to justify a change to the FSAR to document the range of acceptable cracking pressures for the HUT vacuum breakers (1.0 - 3.0 in.-WC), and to document the minimum required vacuum breaker flow capacity.

The recommended FSAR changes were provided in Attachment 1 to the evaluation. Attachment 2 provided the vendor technical data sheet for the HUT vacuum breakers. Attachment 3 provided the vendor calculation of actual cracking pressure.

**Safety Evaluation:**

This evaluation concluded that the CVCS HUTs are in compliance with the Turkey Point FSAR and the evaluated changes for correctness and clarification did not change the operation, function, or design bases of any structure, system, or component important to safety. Consequently, the FSAR changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Thus, prior NRC approval was not required to implement the documentation changes.

**SAFETY EVALUATION PTN-ENG-SECS-00-069**
REVISION 0

UNIT : 4
APPROVAL DATE : 09/19/00

## SAFETY EVALUATION FOR THE USE OF UNITS 1 & 2 TURBINE GANTRY CRANE ON THE UNITS 3 & 4 TURBINE OPERATING DECK


### Summary:

This safety evaluation addressed the use of the Turkey Point Units 1 and 2 turbine gantry crane on the Units 3 and 4 turbine operating deck during the Unit 4 Cycle 19 refueling outage. The proposed crane utilization was necessary to facilitate overhaul of the Unit 4 turbines and turbine support equipment and reduce the usage demand on the Units 3 and 4 gantry crane. This evaluation provided a review of NUREG-0612 and FPL licensing commitments related to heavy load handling, an assessment of the increase in turbine building loads due to simultaneous operation of the two gantry cranes, and evaluation of the impact on physical plant security when the barrier between Units 2 and 3 is breached. Administrative controls were established to ensure compliance with NUREG-0612 in lieu of making physical modifications to the Units 1 and 2 crane.


### Safety Evaluation:

The safety evaluation concluded that the Units 1 and 2 turbine gantry crane can be safely used on Units 3 and 4 provided that all of the restrictions and requirements identified within the evaluation were implemented. The evaluation further concluded that the identified restrictions and requirements would ensure that the proposed activity would have no adverse effects on plant operation, and would not compromise the safety and licensing bases for Units 3 and 4. Consequently, the actions and limitations identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions and limitations identified within this evaluation.

## SAFETY EVALUATION PTN-ENG-SEMS-00-071
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/24/00

## SAFETY EVALUATION FOR FIRE HAZARD ANALYSIS FSAR CHANGES


**Summary:**

The purpose of this safety evaluation was to address appropriate FSAR changes related to the presence of carpeting in the control room. During routine design activities, it was noted that carpeting in the control room was not addressed for combustible loading in the Fire Hazards Analysis section of the FSAR (Appendix 9.6A, Subsection 4.MM.2). Since carpeting was already provided for by design and since combustible loading is not used in the design basis for fire protection, the omission was not considered to be an operability concern.

The evaluation established the combustible loading for the carpeting based on the quantity installed and a conservative estimate of its fuel content.

The recommended FSAR changes were provided in an Attachment 1 to the evaluation. The changes reflect the presence of carpet in the control room, the material quantity installed, and its combustible loading.


**Safety Evaluation:**

The evaluation concluded that the identified FSAR discrepancy did not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, did not constitute an unreviewed safety question and did not require a change to the technical specifications. Consequently, pursuant to the requirements of 10 CFR 50.59, the resulting updates to the Turkey Point FSAR for correctness and clarification did not require NRC approval prior to implementation.

## SAFETY EVALUATION PTN-ENG-SENS-00-072
REVISION 0

| | | |
|---|---|---|
| UNIT | : | 4 |
| APPROVAL DATE | : | 09/22/00 |

## IN SITU HYDROSTATIC TESTING OF STEAM GENERATOR TUBE FLAWS

**Summary:**

This evaluation was prepared as a contingency measure in the event that in situ hydrostatic test of a steam generator (S/G) tube was required during the Turkey Point Unit 4 Cycle 19 refueling outage. Historically, pressure testing has been performed in the laboratory, thus requiring removal of the degraded tubes from the S/G to demonstrate that they can withstand the pressure requirements of draft Regulatory Guide (RG) 1.121, *"Basis for Plugging Degraded PWR Steam Generator Tubes."* More recently equipment has been developed to pressure test tubes in situ. Guidelines for in situ testing of S/G tubes have been developed by the industry and formally published by EPRI. They provide a guide to develop and justify plant specific in situ pressure test procedures as a means to assess the structural integrity of a degraded tube. One approach to validate the results of the examination techniques is to demonstrate via pressure testing that defective tubes can sustain the pressure requirements of draft RG without bursting or leaking beyond analyzed limits. This evaluation provides the pressures and plant restrictions/criteria for the in situ pressure testing, as well as the basis for the test pressures to be used and demonstrates that the test pressures meet the criteria of RG 1.121 and that they are safe to conduct in accordance with 10 CFR 50.59 requirements.

**Safety Evaluation:**

This evaluation examined the test equipment and the methods used to test the various types of flaw indications, the test pressure and hold time, the impact of testing on plant operation, failure modes and effects, plant operating restrictions during testing, and applicable compensatory measures. The evaluation concluded that the proposed testing approach and test equipment had no adverse impact on plant safety or plant operations, and therefore, did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to perform the subject pressure test.

## SAFETY EVALUATION PTN-ENG-SENS-00-073
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/15/00

## SAFETY EVALUATION FOR USE OF A FREEZE SEAL ON THE AFW PUMP MINIMUM FLOW RECIRCULATION PIPING


**Summary:**

This safety evaluation was prepared to assess the performance and use of a freeze seal(s) when conducting repairs on the auxiliary feedwater (AFW) pump minimum flow recirculation piping. It was generated in response to an identified through-wall defect located downstream of the "A" AFW pump minimum flow restricting orifice, in the weld area. Non-destructive examinations revealed that the defect was caused by erosion. The repair option of choice involved replacement of the affected piping segment with an erosion resistant material. Suitable isolation valves existed downstream of the restricting orifice to provide an isolation boundary while the piping repairs were made but the isolation boundary affected operability of all three AFW pumps. A freeze seal was required to isolate the affected piping segment during the repair process and maintain operability of the remaining pumps. The controlled plant procedure governing freeze seal application was referenced in the evaluation, and contingency plans were established to restore pressure boundary integrity for the open system upon indication of freeze seal deterioration.

This evaluation was written such that it can be applied to the repair of a minimum flow recirculation line on any AFW pump during operating Modes 1, 2, or 3.

A sketch detailing acceptable freeze seal locations was provided as Attachment 1 to the evaluation.


**Safety Evaluation:**

The freeze seals were relied on to perform an AFW system boundary function during the short repair duration. The strict controls imposed on the freeze seal process, the contingency measures, relatively low pressure of the contained fluid, and small size of the piping opening ensured that all AFW safety functions would remain unimpaired throughout the installation. Based on the precautions identified in this safety evaluation, it was concluded that the freeze seal(s) could be performed, and that the activity did not involve an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SEMS-00-074**
REVISION 0

| | | |
|---|---|---|
| UNIT | : | 3 & 4 |
| APPROVAL DATE | : | 08/17/00 |

## SAFETY EVALUATION FOR SI BACKLEAKAGE ISOLATION


**Summary:**

This safety evaluation provided the basis to temporarily isolate a potential back-leakage flow path from the Unit 4 safety injection (SI) accumulators to the Unit 4 high head safety injection (HHSI) pumps. The flow path was suspected of being a source of nitrogen intrusion to the HHSI system, and was anticipated to remain isolated until the leaking valves were repaired during the Unit 4 Cycle 19 refueling outage. The proposed method of isolation was to close valve 4-895P. Although this change would block the relief flow path from RV-4-857, the evaluation demonstrated that adequate thermal relief protection would be provided by RV-3-857 during those modes of plant operation in which the Unit 3 cross-tie was open. The safety evaluation also reviewed the conditions under which the Unit 3 cross-tie could be isolated. In each case, the evaluation demonstrated that pressure boundary integrity would be maintained by virtue of the ultimate pressure capacity of the installed piping.

The temporary closure of valve 4-895P did not change the plant configuration or method of operation as described in the FSAR because valve manipulation is a normal operating function and the alignment did not undermine the HHSI system pressure boundary, or the capability to perform its safety-related functions. The safety evaluation required that the normal alignment be restored prior to entering Mode 4 during unit startup from the Cycle 19 refueling outage.


**Safety Evaluation:**

This evaluation concluded that temporary isolation of the RV-4-857 discharge path would not alter the flow delivery functions of the HHSI system during postulated accidents. The review also demonstrated that the pressure boundary integrity of the system would not be impacted by virtue of the ultimate pressure capacity of the installed piping, or the alternate relief protection provided by RV-3-857, if available. It was concluded that the actions and temporary plant conditions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation of the actions or temporary plant conditions identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SEMS-00-077**
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 09/26/00

## SAFETY EVALUATION FOR USE OF CARBOHYDRAZIDE IN STEAM GENERATOR SECONDARY SIDE

### Summary:

This safety evaluation provided the technical justification to use carbohydrazide to scavenge oxygen in the secondary system at ambient temperatures. Control of dissolved oxygen is a primary part of steam generator corrosion control. Multiple approaches are available to control the level of dissolved oxygen in the secondary system including control of oxygen within makeup sources, mechanical deaeration and chemical scavenging. Turkey Point uses both mechanical and chemical controls to effectively remove dissolved oxygen during normal power operation. However, at cold shutdown conditions, those control techniques are either not available or are ineffective. Carbohydrazide has been confirmed by both testing and industry experience to be effective for oxygen scavenging at ambient temperatures. This evaluation assessed the use of this additional compound in the secondary side of the steam generators during wet lay-up conditions. It addressed various aspects associated with the qualification of a new chemical for use in the steam generators including chemical and material compatibility, OSHA and environmental exposure concerns, and application requirements.

### Safety Evaluation:

This safety evaluation demonstrated that the storage, processing and delivery of a carbohydrazide solution to the secondary system is similar to that of existing secondary system water treatment chemicals. No new hazards are created by the alternate treatment scheme and there is no reduction in system piping or component reliability. The actions and procedural changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required to implement the actions or changes identified within this evaluation.

**SAFETY EVALUATION PTN-ENG-SECS-00-082**
REVISION 0

UNIT : 4
APPROVAL DATE : 10/01/00

# SAFETY EVALUATION FOR TEMPORARY REMOVAL OF THE CCW ROOM MISSILE BARRIER

## Summary:

This safety evaluation provided the technical and licensing justification to permit temporary removal of the tornado missile barrier above the Unit 4 component cooling water (CCW) room. The CCW rooms are protected against tornado missiles by steel grating and reinforced concrete walls. Temporary removal of the steel grating was required during the Cycle 19 refueling outage to facilitate implementation of the modifications addressed in PC/M 00-023. In order to maintain the CCW system within its design basis, administrative controls were established to replace the grating panels in the event that a Tornado Watch was established for areas surrounding the plant.

## Safety Evaluation:

The evaluation concluded that the temporary condition did not represent a change to the facility as described in the FSAR provided that the precautions and limitations identified in the evaluation were complied with. It further concluded that the specified precautions and limitations would not reduce the existing level of protection provided against tornado missiles. Consequently, pursuant to 10 CFR 50.59, the actions and limitations provided in the evaluation did not constitute an unreviewed safety question, or require a change to the plant technical specifications. Therefore, prior NRC approval was not required to perform the evaluated activity.

## SAFETY EVALUATION PTN-ENG-SENS-00-083
REVISION 0

UNIT : 3
APPROVAL DATE : 10/20/00

## SAFETY EVALUATION FOR INSTALLATION OF A REMOVABLE STEM LOCKING DEVICE ON VALVE HCV-3-758


**Summary:**

This safety evaluation analyzes the impact on plant safety and operation associated with the temporary installation of a stem lock on valve HCV-3-758 during Cycle 18 operation. The locking device was installed in response to a failure of the shaft key on the associated Unit 4 valve that occurred during the Unit 4 Cycle 19 refueling outage. Although the shaft key on valve HCV-3-758 appeared to be intact, the stem lock was intended to provide additional assurance that valve HCV-3-758 will remain in its pre-set open position during normal plant operation to support the low head safety injection function of the residual heat removal (RHR) system. The locking device is designed to be removable to support manual operation of the RHR system during normal and safe shutdowns. Since removal of the stem lock represents an additional action that must be performed by the attendant Operator to place the RHR system in service, this safety evaluation uses the criteria of 10 CFR 50.59 to determine if the additional local manual action represents an unreviewed safety question or requires a change to the plant technical specifications.


**Safety Evaluation:**

This evaluation determined that the temporary installation of a removable locking device on valve HCV-3-758 did not invalidate any licensing or design basis requirements for the RHR system. It concluded that the activity did not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, did not constitute an unreviewed safety question, or require a change to the plant technical specifications. Therefore, prior NRC approval was not required to complete the proposed activity.

SECTION 3

RELOAD SAFETY EVALUATIONS

## PLANT CHANGE/MODIFICATION 99-015

| | | |
|---|---|---|
| UNIT | : | 3 |
| TURN OVER DATE | : | 08/30/00 |

## TURKEY POINT UNIT 3 CYCLE 18 RELOAD DESIGN

**Summary:**

This Engineering Package provided the reload core design for the Turkey Point Unit 3 Cycle 18 reload. The primary design change to the core for Cycle 18 was the replacement of 69 irradiated assemblies with 60 fresh Optimized Fuel Assembly (OFA) Region 20 fuel assemblies and 9 Region 16 twice-burned assemblies. Similar to past reloads, these fresh assemblies were all Debris Resistant Fuel Assemblies (DRFA) and all contained a nominal 6-inch axial blanket of natural $UO_2$ pellets at both the top and bottom of the fuel stack. The maximum fuel enrichment was 4.4 w/o. Cycle 18 is the first cycle in which Unit 3 is operating with annular natural uranium pellets in the axial blankets of the new assemblies and with no discrete burnable poison inserts.

Fuel assembly JJ-29 was reconstituted during the refueling outage and re-inserted in the core for Cycle 18 with one replacement stainless steel rod.

Cross core fuel bundle shuffles were utilized in the Cycle 18 loading pattern to minimize potential power asymmetries. The fuel was arranged in a low leakage pattern with no significant differences between the Cycle 17 and Cycle 18 patterns.

**Safety Evaluation:**

The Unit 3 Cycle 18 reload core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric Corporation. The Cycle 18 reload core design met all applicable design criteria, appropriate licensing bases, and the requirements of plant technical specifications. The minor design modifications to fuel assemblies in this reload did not affect applicable design criteria and did not increase the radiological consequences of any accident previously evaluated in the SAR. These changes had no impact on fuel rod performance, dimensional stability or core operating limits. The Cycle 18 core reload did not have any adverse effect on plant safety or plant operations. Consequently, the Cycle 18 core reload package did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

## PLANT CHANGE/MODIFICATION 00-011

| | | |
|---|---|---|
| UNIT | : | 4 |
| TURN OVER DATE | : | 12/31/00 |

## TURKEY POINT UNIT 4 CYCLE 19 RELOAD DESIGN

Summary:

This Engineering Package provided the reload core design for the Turkey Point Unit 4 Cycle 19 reload. The primary design change to the core for Cycle 19 was the replacement of 56 irradiated assemblies with 56 fresh Optimized Fuel Assembly (OFA) Region 21 fuel assemblies. Similar to past reloads, these fresh assemblies were all Debris Resistant Fuel Assemblies (DRFA) and all contained a nominal 6-inch axial blanket of natural $UO_2$ pellets at both the top and bottom of the fuel stack. The maximum fuel enrichment was 4.4 w/o which is consistent with the previous cycle.

Minor changes were made to the Region 21 fuel assembly design. These changes included the use of bead blasted alloy 600 top nozzle spring screws and the use of a replacement removable top nozzle for use in a fuel assembly top nozzle that was repaired during the refueling outage. None of these manufacturing-related design changes had any impact on fuel performance.

Cross core fuel bundle shuffles were utilized to minimize potential power asymmetries and a low leakage loading pattern was utilized similar to past core designs.

**Safety Evaluation:**

The Unit 4 Cycle 19 reload core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric Corporation. The Cycle 19 reload core design met all applicable design criteria, appropriate licensing bases, and the requirements of plant technical specifications. The minor design modifications to fuel assemblies in this reload did not affect applicable design criteria and did not increase the radiological consequences of any accident previously evaluated in the SAR. These changes had no impact on fuel rod performance, dimensional stability or core operating limits. The Cycle 19 core reload did not have any adverse effect on plant safety or plant operations. Consequently, the Cycle 19 core reload package did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation.

SECTION 4

# 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 their intent to comply with the requirements of Item II.K.3.3 of Enclosure 3 to the Commissioner's letter of May 7, 1980 (Five Additional TMI-2 Related Requirements for Operating Reactors). Pursuant to these requirements, a summary of the power operated relief valve (PORV) actuations that have occurred at Turkey Point during this reporting period is provided below:

### Unit 3

No PORV actuations have occurred on Unit 3 between April 9, 1999 and October 23, 2000.

### Unit 4

No PORV actuations have occurred on Unit 4 between April 9, 1999 and October 23, 2000.

# SECTION 5

# STEAM GENERATOR TUBE INSPECTIONS FOR TURKEY POINT

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

EDDY CURRENT EXAMINATION RESULTS

PLANT: Turkey Point Unit 3

EXAMINATION DATE: March 10, 2000 thru March 15, 2000

| STEAM GENERATOR | TOTAL TUBES INSPECTED | TOTAL TUBES 20%-39% | TOTAL TUBES ≥40%, VOL, Circ. | TUBES PREVENTIVELY PLUGGED (PTP) | TUBES PLUGGED THIS OUTAGE | TOTAL PLUGGED TUBES IN S/G |
|---|---|---|---|---|---|---|
| 3E210A (Bobbin) | 1609 | 5 (1) | 0 | 2 (2) | 2 (2) | See RPC |
| 3E210B (Bobbin) | 1601 | 4 (1) | 0 | 1 (2) | 1 (2) | See RPC |
| 3E210C (Bobbin) | 1627 | 19 (1) | 1 | 1 (2) | 2 (2) | See RPC |
| 3E210A (RPC) | 3194 (4) | 0 | 23 | 0 | 23 (5) | 45 |
| 3E210B (RPC) | 3186 (4) | 0 | 27 | 0 | 27 (5) | 56 |
| 3E210C (RPC) | 3179 (4) | 0 | 14 | 0 | 14 (5) | 51 |

LOCATION OF INDICATIONS
(20% - 100%, VOL & Circ.)

| STEAM GENERATOR | AVB Bars | Tube Supports 1 thru 6 C/L | Tube Supports 1 thru 6 H/L | Freespan 6H thru 6C UBEND | Top of Tubesheet to #1 Support C/L | Top of Tubesheet to #1 Support H/L | Total Indications 20%-39% | Total Indications >40%, VOL, Circ. |
|---|---|---|---|---|---|---|---|---|
| 3E210A (Bobbin) | 7 (3) | 0 | 0 | 0 | 0 | 0 | | 0 |
| 3E210B (Bobbin) | 6 (3) | 0 | 0 | 0 | 0 | 0 | | 0 |
| 3E210C (Bobbin) | 33 (3) | 0 | 0 | 0 | 0 | 0 | | 2 (1) (3) |
| 3E210A (RPC) | 0 | 0 | 0 | 0 | n/a | 24 | n/a | 24 (3) (5) |
| 3E210B (RPC) | 0 | 0 | 0 | 0 | n/a | 29 | n/a | 29 (3) (5) |
| 3E210C (RPC) | 0 | 0 | 0 | 0 | n/a | 16 | n/a | 16 (3) (5) |

Remarks:
(1) Mechanical wear damage at anti-vibration bars (AVB) was depth sized using qualified bobbin coil sizing technique.
(2) Two tubes in 3A, one tube in 3B and one tube in 3C were preventatively plugged for AVB wear progression.
(3) Some tubes may have more than one indication reported.
(4) Includes tubes in the dent, low row U-bend and hot leg TTS expansion transition programs.
(5) Includes volumetric (VOL) and circumfential (Circ.) indications.

DATE : 4/4/00 PREPARED BY : Alonso Montalbano
CSI S/G EDDY CURRENT COORDNIATOR

DATE : 4/10/00 REVIEWED BY : Glenn C. Alexander
CSI INSPECTIONS SUPERVISOR

DATE : 4/16/00 REVIEWED BY : M L Boggess
CSI S/G TECHNICAL SPECIALIST

# PTN-3 S/G "A"

# Indication Report

4/11/00 8:04:56 AM

Page 1 of 1

| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 28 | 59 | AC002 | 0.68 | 0 | P 2 | 23 | | AV2 | 0 |
| 33 | 15 | AC008 | 0.64 | 0 | P 2 | 23 | | AV3 | 0 |
| 33 | 41 | AC002 | 1.57 | 0 | P 2 | 34 | | AV3 | 0 |
| 33 | 41 | AC002 | 0.87 | 0 | P 2 | 26 | | AV1 | 0 |
| 37 | 47 | AC002 | 1.01 | 0 | P 2 | 28 | | AV3 | 0 |
| 38 | 45 | AC002 | 0.79 | 0 | P 2 | 24 | | AV3 | 0 |
| 38 | 45 | AC002 | 1.88 | 0 | P 2 | 37 | | AV2 | 0 |

TOTAL INDICATIONS: 7

TOTAL TUBES: 5

# PTN-3 S/G "B"

# Indication Report

4/11/00 8:10:05 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 32 | 34 | BC001 | 1.17 | 0 | P2 | 27 | | AV3 | 0 |
| 32 | 34 | BC001 | 0.8 | 0 | P2 | 21 | | AV1 | 0 |
| 34 | 46 | BC001 | 1.71 | 0 | P2 | 34 | | AV3 | 0 |
| 34 | 46 | BC001 | 1.46 | 0 | P2 | 31 | | AV2 | 0 |
| 34 | 51 | BC001 | 1.41 | 0 | P2 | 31 | | AV2 | 0 |
| 34 | 53 | BC001 | 0.75 | 0 | P2 | 20 | | AV2 | 0 |

TOTAL INDICATIONS: 6

TOTAL TUBES: 4

# PTN-3 S/G "C"

# Indication Report

4/11/00 8:04:18 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 23 | 45 | CC001 | 0.39 | 0 | P 2 | 20 | | AV3 | 0 |
| 24 | 63 | CC001 | 0.52 | 0 | P 2 | 24 | | AV3 | 0 |
| 25 | 62 | CC001 | 0.47 | 0 | P 2 | 23 | | AV3 | 0 |
| 26 | 58 | CC001 | 0.7 | 0 | P 2 | 29 | | AV2 | 0 |
| 26 | 58 | CC001 | 0.4 | 0 | P 2 | 20 | | AV1 | 0 |
| 28 | 48 | CC001 | 0.5 | 0 | P 2 | 23 | | AV2 | 0 |
| 30 | 31 | CC001 | 0.46 | 0 | P 2 | 22 | | AV2 | 0 |
| 30 | 31 | CC001 | 0.5 | 0 | P 2 | 23 | | AV3 | 0 |
| 30 | 61 | CC001 | 0.59 | 0 | P 2 | 26 | | AV2 | 0 |
| 33 | 31 | CC001 | 0.46 | 0 | P 2 | 22 | | AV3 | 0 |
| 33 | 43 | CC002 | 0.55 | 0 | P 2 | 26 | | AV3 | 0 |
| 33 | 43 | CC002 | 0.37 | 0 | P 2 | 20 | | AV2 | -0.18 |
| 34 | 31 | CC001 | 0.69 | 0 | P 2 | 29 | | AV3 | 0 |
| 34 | 31 | CC001 | 0.56 | 0 | P 2 | 25 | | AV2 | 0 |
| 34 | 41 | CC002 | 0.62 | 0 | P 2 | 28 | | AV2 | 0 |
| 34 | 41 | CC002 | 0.76 | 0 | P 2 | 31 | | AV3 | 0 |
| 34 | 41 | CC002 | 0.78 | 0 | P 2 | 32 | | AV4 | 0 |
| 34 | 41 | CC002 | 0.78 | 0 | P 2 | 32 | | AV1 | 0 |
| 34 | 44 | CC002 | 0.43 | 0 | P 2 | 23 | | AV3 | 0 |
| 35 | 36 | CC002 | 0.39 | 0 | P 2 | 21 | | AV2 | 0 |
| 35 | 36 | CC002 | 0.53 | 0 | P 2 | 26 | | AV3 | 0 |

| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 35 | 43 | CC002 | 0.9 | 0 | P2 | 34 | | AV2 | 0 |
| 35 | 43 | CC002 | 1.2 | 0 | P2 | 38 | | AV3 | 0 |
| 35 | 43 | CC002 | 0.74 | 0 | P2 | 31 | | AV4 | 0 |
| 35 | 43 | CC002 | 0.42 | 0 | P2 | 22 | | AV1 | 0 |
| 35 | 44 | CC002 | 1.67 | 0 | P2 | 42 | | AV2 | 0 |
| 35 | 44 | CC002 | 1.68 | 0 | P2 | 43 | | AV3 | 0 |
| 35 | 44 | CC002 | 0.53 | 0 | P2 | 26 | | AV4 | 0 |
| 35 | 49 | CC002 | 0.41 | 0 | P2 | 22 | | AV4 | 0 |
| 38 | 65 | CC002 | 0.51 | 0 | P2 | 25 | | AV2 | 0 |
| 38 | 65 | CC002 | 0.55 | 0 | P2 | 26 | | AV4 | 0 |
| 38 | 71 | CC002 | 0.59 | 0 | P2 | 27 | | AV3 | 0 |
| 40 | 25 | CC001 | 0.39 | 0 | P2 | 20 | | AV2 | 0.03 |

**TOTAL INDICATIONS:** 33

**TOTAL TUBES:** 19

# PTN-3 S/G "A"

# Pluggable Report

4/11/00 8:08:18 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 3 | 80 | AH064 | 1.07 | 19 | 2 | 0 | CSI | TSH | -0.08 |
| 10 | 31 | AH039 | 0.78 | 21 | 2 | 0 | CSI | TSH | -0.15 |
| 16 | 64 | AH014 | 1.02 | 20 | 2 | 0 | CSI | TSH | -0.09 |
| 17 | 15 | AH069 | 0.22 | 38 | 2 | 0 | CSI | TSH | 0.05 |
| 17 | 33 | AH054 | 0.12 | 95 | 2 | 0 | CSI | TSH | 0.15 |
| 18 | 83 | AH060 | 0.15 | 101 | P 1 | 0 | VOL | TSH | 0.1 |
| 18 | 84 | AH059 | 0.28 | 100 | P 1 | 0 | VOL | TSH | 0.16 |
| 19 | 84 | AH060 | 0.3 | 97 | P 1 | 0 | VOL | TSH | 0.42 |
| 19 | 84 | AH060 | 0.27 | 88 | P 1 | 0 | VOL | TSH | 0.94 |
| 21 | 87 | AH059 | 0.08 | 86 | P 1 | 0 | VOL | TSH | 0.69 |
| 28 | 75 | AH057 | 0.39 | 93 | P 1 | 0 | VOL | TSH | 0.15 |
| 29 | 75 | AH058 | 0.18 | 84 | P 1 | 0 | VOL | TSH | 0.05 |
| 30 | 65 | AH057 | 0.07 | 114 | P 1 | 0 | VOL | TSH | 0.24 |
| 31 | 77 | AH058 | 0.14 | 130 | P 1 | 0 | VOL | TSH | 0.1 |
| 32 | 23 | AH043 | 1.09 | 18 | 2 | 0 | CSI | TSH | -0.05 |
| 32 | 63 | AH055 | 0.14 | 140 | 2 | 0 | CSI | TSH | 0.05 |
| 32 | 64 | AH010 | 0.08 | 123 | 2 | 0 | CSI | TSH | -0.01 |
| 33 | 35 | AH045 | 0.07 | 91 | 2 | 0 | CSI | TSH | -0.02 |
| 33 | 41 | AC002 | 1.57 | 0 | P 2 | | PTP | AV3 | 0 |
| 33 | 78 | AH061 | 0.11 | 104 | P 1 | 0 | VOL | TSH | 0.75 |
| 34 | 25 | AH045 | 1.34 | 16 | 2 | 0 | CSI | TSH | -0.08 |
| 35 | 65 | AH062 | 0.09 | 141 | P 1 | 0 | VOL | TSH | 0.96 |
| 36 | 69 | AH062 | 0.09 | 122 | P 1 | 0 | VOL | TSH | 0.21 |

| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 38 | 45 | AC002 | 1.88 | 0 | P2 | | PTP | AV2 | 0 |
| 38 | 66 | AH061 | 0.09 | 96 | P1 | 0 | VOL | TSH | 0.23 |
| 39 | 67 | AH062 | 0.07 | 112 | P1 | 0 | VOL | TSH | -0.05 |

TOTAL INDICATIONS: 26

TOTAL TUBES: 25

# PTN-3 S/G "B"

# Pluggable Report

4/11/00 9:38:00 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 14 | BH025 | 0.93 | 24 | 2 | 0 | CSI | TSH | -0.28 |
| 7 | 92 | BH053 | 0.07 | 102 | 2 | 0 | VOL | TSH | 0.57 |
| 15 | 17 | BH073 | 0.75 | 26 | 2 | 0 | CSI | TSH | -0.06 |
| 19 | 10 | BH004 | 0.13 | 99 | 2 | 0 | VOL | TSH | 0.24 |
| 19 | 12 | BH030 | 0.21 | 106 | 2 | 0 | VOL | TSH | 0.54 |
| 19 | 13 | BH028 | 0.61 | 90 | 2 | 0 | VOL | TSH | 0.25 |
| 19 | 14 | BH004 | 0.29 | 132 | 2 | 0 | VOL | TSH | 0.29 |
| 20 | 10 | BH061 | 0.11 | 88 | 2 | 0 | VOL | TSH | 0.03 |
| 20 | 12 | BH028 | 0.13 | 125 | 2 | 0 | VOL | TSH | 0.21 |
| 20 | 13 | BH030 | 0.19 | 100 | 2 | 0 | VOL | TSH | 0.03 |
| 21 | 56 | BH008 | 0.04 | 59 | 2 | 0 | VOL | TSH | 0.43 |
| 22 | 53 | BH070 | 0.07 | 51 | P1 | 0 | VOL | TSH | 0.6 |
| 23 | 7 | BH028 | 0.09 | 104 | 2 | 0 | VOL | TSH | 0.58 |
| 25 | 34 | BH004 | 0.08 | 99 | 2 | 0 | VOL | TSH | 0.2 |
| 26 | 71 | BH065 | 0.26 | 123 | P1 | 0 | VOL | TSH | 0.09 |
| 33 | 70 | BH044 | 0.11 | 97 | 2 | 0 | CSI | TSH | -0.06 |
| 34 | 46 | BC001 | 1.71 | 0 | P2 | | PTP | AV3 | 0 |
| 34 | 57 | BH043 | 0.33 | 90 | P1 | 0 | VOL | TSH | 0.1 |
| 37 | 46 | BH037 | 0.24 | 113 | 2 | 0 | VOL | TSH | 0.04 |
| 38 | 39 | BH037 | 0.5 | 91 | 2 | 0 | CSI | TSH | 0.05 |
| 38 | 39 | BH037 | 0.62 | 107 | 2 | 0 | CSI | TSH | -0.06 |
| 38 | 45 | BH037 | 0.23 | 107 | 2 | 0 | CSI | TSH | 0.06 |
| 38 | 45 | BH037 | 0.15 | 107 | 2 | 0 | CSI | TSH | 0.2 |

| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 38 | 46 | BH038 | 0.19 | 99 | 2 | 0 | VOL | TSH | 0.59 |
| 39 | 59 | BH044 | 0.08 | 117 | P 1 | 0 | VOL | TSH | 0.19 |
| 41 | 43 | BH059 | 0.12 | 77 | 2 | 0 | VOL | TSH | 0.04 |
| 41 | 65 | BH043 | 0.12 | 89 | P 1 | 0 | VOL | TSH | 0.63 |
| 43 | 33 | BH059 | 0.15 | 69 | 2 | 0 | VOL | TSH | 0.14 |
| 44 | 42 | BH006 | 0.1 | 130 | 2 | 0 | VOL | TSH | 0.4 |
| 45 | 47 | BH040 | 0.1 | 67 | P 1 | 0 | VOL | TSH | 0.64 |

TOTAL INDICATIONS: 30

TOTAL TUBES: 28

# PTN-3 S/G "C"

## Pluggable Report

4/11/00 8:09:25 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 20 | CH068 | 1.29 | 21 | 2 | 0 | CSI | TSH | -0.12 |
| 3 | 46 | CH068 | 0.9 | 22 | 2 | 0 | CSI | TSH | -0.05 |
| 7 | 3 | CH082 | 0.15 | 113 | P1 | 0 | VOL | TSH | 0.09 |
| 15 | 44 | CH068 | 0.62 | 34 | 2 | 0 | CSI | TSH | 0.03 |
| 22 | 7 | CH069 | 0.29 | 119 | 2 | 0 | VOL | TSH | 0.55 |
| 23 | 7 | CH080 | 0.24 | 112 | P1 | 0 | VOL | TSH | 0.59 |
| 30 | 69 | CH061 | 1.17 | 14 | 2 | 0 | CSI | TSH | -0.03 |
| 31 | 24 | CH047 | 0.14 | 111 | 2 | 0 | CSI | TSH | 0.16 |
| 34 | 40 | CH075 | 0.61 | 13 | 2 | 0 | CSI | TSH | -0.08 |
| 34 | 66 | CH077 | 0.08 | 128 | P1 | 0 | VOL | TSH | 0.23 |
| 35 | 43 | CC002 | 1.2 | 0 | P2 | | PTP | AV3 | 0 |
| 35 | 44 | CC002 | 1.68 | 0 | P2 | 43 | | AV3 | 0 |
| 35 | 44 | CC002 | 1.67 | 0 | P2 | 42 | | AV2 | 0 |
| 36 | 74 | CH076 | 0.38 | 13 | 2 | 0 | CSI | TSH | -0.07 |
| 39 | 49 | CH079 | 0.36 | 125 | P1 | 0 | VOL | TSH | -0.01 |
| 40 | 49 | CH072 | 0.17 | 87 | 2 | 0 | CSI | TSH | 0.06 |
| 40 | 49 | CH072 | 0.22 | 107 | 2 | 0 | CSI | TSH | -0.08 |
| 45 | 49 | CH072 | 0.06 | 58 | 2 | 0 | VOL | TSH | 1.37 |
| 45 | 49 | CH072 | 0.04 | 73 | 2 | 0 | VOL | TSH | 2.89 |

TOTAL INDICATIONS: 19

TOTAL TUBES: 16

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

EDDY CURRENT EXAMINATION RESULTS

PLANT: Turkey Point Unit 4

EXAMINATION DATE: October 4, 2000 thru October 9, 2000

| STEAM GENERATOR | TOTAL TUBES INSPECTED | TOTAL TUBES 20%-39% | TOTAL TUBES ≥40%, PIT & VOL | TUBES PREVENTIVELY PLUGGED (PTP) | TUBES PLUGGED THIS OUTAGE | TOTAL PLUGGED TUBES IN S/G |
|---|---|---|---|---|---|---|
| 4E210A (Bobbin) | 1602 | 0 | 0 | 0 | 0 | See RPC |
| 4E210B (Bobbin) | 1604 | 1 (1) | 0 | 1 (2) | 1 (2) | See RPC |
| 4E210C (Bobbin) | 1607 | 3 (1) | 0 | 1 (3) | 1 (3) | See RPC |
| 4E210A (RPC) | 3242 (4) | 0 | 2 | 1 (5) | 3 (6) | 19 |
| 4E210B (RPC) | 3247 (4) | 0 | 4 | 0 | 4 (6) | 13 |
| 4E210C (RPC) | 3254(4) | 0 | 1 | 0 | 1 (6) | 11 |

LOCATION OF INDICATIONS
(20% - 100%, PIT & VOL)

| STEAM GENERATOR | AVB Bars | Tube Supports 1 thru 6 C/L | Tube Supports 1 thru 6 H/L | Freespan 6H thru 6C UBEND | Top of Tubesheet to #1 Support C/L | Top of Tubesheet to #1 Support H/L | Total Indications 20%-39% | Total Indications ≥40%, PIT & VOL |
|---|---|---|---|---|---|---|---|---|
| 4E210A (Bobbin) | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 4E210B (Bobbin) | 1 (1) | 0 | 0 | 0 | 0 | 0 | 1 | 0 |
| 4E210C (Bobbin) | 3 (1) | 0 | 0 | 0 | 0 | 0 | 3 | 0 |
| 4E210A (RPC) | 0 | n/a | n/a | 0 | n/a | 2 | n/a | 2 |
| 4E210B (RPC) | 0 | n/a | n/a | 0 | n/a | 4 | n/a | 4 |
| 4E210C (RPC) | 0 | n/a | n/a | 0 | n/a | 1 | n/a | 1 |

Remarks:
(1) Mechanical wear damage at anti-vibration bars (AVB) was depth sized using qualified bobbin coil sizing technique.
(2) One tube in 4B was preventatively plugged due to minor wear at the hot leg baffle plate.
(3) One tube in 4C was preventatively plugged for AVB wear progression.
(4) Includes tubes in the dent, low row U-bend and hot leg TTS expansion transition programs.
(5) One tube in 4A was preventatively plugged due to permeability in the hot leg expansion transition area.
(6) Includes volumetric (VOL) and pit (PIT) like indications.

# PTN-4 S/G "B

# Indication Report

10/16/00 9:40:28 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 34 | 46 | BC009 | 0.55 | 0 | P 2 | 20 | | AV2 | -0.35 |

TOTAL INDICATIONS: 1

TOTAL TUBES: 1

# PTN-4 S/G "C

# Indication Report

10/16/00 9:43:24 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 13 | 43 | CC006 | 1.07 | 0 | P2 | 36 | | AV3 | -0.79 |
| 32 | 70 | CC012 | 0.61 | 0 | P2 | 24 | | AV1 | -0.09 |
| 35 | 31 | CC013 | 0.67 | 0 | P2 | 28 | | AV2 | 0.15 |

TOTAL INDICATIONS: 3

TOTAL TUBES: 3

# PTN-4 S/G "A

## Pluggable Indications

10/16/00 9:37:17 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 12 | 25 | AH009 | 0 | 0 | | 0 | PTP | | 0 |
| 26 | 80 | AH028 | 0.16 | 92 | P 1 | 0 | PIT | TSH | 2.27 |
| 33 | 73 | AH029 | 0.41 | 114 | P 1 | 0 | VOL | TSH | 0.17 |

TOTAL INDICATIONS: 3

TOTAL TUBES: 3

# PTN-4 S/G "B

## Pluggable Indications

10/16/00 9:41:42 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 2 | 90 | BH056 | 0 | 0 | P 4 | 0 | PTP | BAH | -0.46 |
| 20 | 80 | 2H025 | 0.23 | 114 | P 1 | 0 | PIT | TSH | 0 |
| 21 | 80 | BH024 | 0.08 | 91 | P 1 | 0 | PIT | TSH | 0.05 |
| 29 | 62 | BH007 | 0.07 | 84 | P 1 | 0 | PIT | TSH | 0.12 |
| 43 | 51 | BH023 | 0.25 | 107 | P 1 | 0 | PIT | TSH | -0.04 |

TOTAL INDICATIONS: 5

TOTAL TUBES: 5

# PTN-4 S/G "C

## Pluggable Indications

10/16/00 9:43:17 AM



| ROW | COL | CAL | VOLTS | DEG | CH | % | IND | SUPPORT | INCHES |
|---|---|---|---|---|---|---|---|---|---|
| 3 | 91 | CH011 | 0.86 | 81 | P 1 | 0 | PIT | TSH | 0 |
| 13 | 43 | CH035 | 0 | 0 | | 0 | PTP | AV3 | -0.79 |

TOTAL INDICATIONS: 2

TOTAL TUBES: 2