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 8, 1996 THROUGH OCTOBER 13, 1997

INTRODUCTION

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

i) changes in the facility as described in the SAR

ii) changes in procedures as described in the SAR, and

iii) tests and experiments not described in the SAR,

which are conducted without prior Commission approval, be reported to the Commission for the same period as required by 50.71(e) for the Turkey Point UFSAR update. This report is intended to meet this requirement for the period covering April 8, 1996, through October 13, 1997.

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 section summarizes 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 94-141

UNIT : 3 & 4
TURN OVER DATE : 06/06/96

BORIC ACID EVAPORATORS AND GAS STRIPPER ABANDONMENT

Summary:

This Engineering Package provided the design change documentation required for abandonment of the boric acid evaporators and gas stripper equipment. The original purpose of this equipment was to process reactor coolant effluent in order to conserve boric acid and reduce the processing requirements placed on the liquid radioactive waste disposal system. The boric acid evaporators and gas strippers were designed to receive borated radioactive effluent from numerous sources including excess letdown during startup, reactor coolant loop drains, pressurizer relief tank, and clean radioactive drains. The output of the ion exchange, gas stripping, and evaporation process was separation and reclamation of boric acid and primary water. By the mid-1980s, this method of liquid waste disposal had proven too costly to operate and maintain. Presently, all radioactive liquids requiring cleanup are processed by the waste disposal portable demineralizer skid located in the Radwaste Building, and released to the circulating water system. Spent resins are transferred to shipping containers, dewatered, and sent off-site for final processing and disposal.

All electrical and mechanical equipment associated with the evaporators and gas strippers were abandoned in-place. Power feeds were isolated by lifting leads and de-energizing breakers. The various system isolation valves, including the component cooling water valves to the boric acid evaporators, were previously isolated. A circular plate was installed in a flange upstream of valves CV-*-6598 and CV-*-6599 to prevent potential back leakage from the waste disposal vent header from entering the gas stripper package.

Safety Evaluation:

The boric acid evaporators and gas strippers did not serve any safety related functions and were not required to support safe shutdown of the plant. The in-place abandonment of this equipment had no adverse impact on plant safety or plant operations. As demonstrated in the Engineering Package, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-063

UNIT	:	4
TURN OVER DATE	:	12/02/96

SJAE SPING MONITOR DRYER ENHANCEMENTS

Summary:

This Engineering Package replaced the chiller unit used to condense and remove moisture from the steam jet air ejector SPING sample stream because it was corroded beyond repair. The steam jet air ejector SPING monitors the condenser exhaust gases for particulate, iodine, and noble gas radioactivity as required by Regulatory Guide 1.97. The replacement system utilized a moisture separator tank to remove entrained liquid from the sample stream, and heat tracing to raise the temperature of the sample prior to entering the SPING unit to prevent condensation. The moisture separator tank was installed on the mezzanine deck of the Turbine Building, next to the SPING unit.

The heat tracing and thermal insulation was designed to preclude condensation and maintain process temperatures below the maximum value specified by the SPING vendor. The heat tracing was not designed to meet seismic criteria because any credible failure of the heat tracing would not damage any safety related equipment. Electrical power for the heat tracing circuit was taken from the non-vital side of MCC 4A which has no impact on station battery loads, emergency diesel generator loads, or emergency diesel generator load sequencing.

Safety Evaluation:

The steam jet air ejector SPING monitor does not perform any nuclear safety related functions. However, to preclude the possibility of any adverse interactions with safety related structures, systems and components, the mounting and relocation of drain and sample line tubing, conduit, and associated structural supports, were evaluated in accordance with seismic criteria contained in the UFSAR. Tubing and supports located above the operating deck of the Turbine Building were also evaluated for applicable hurricane wind loads in accordance with UFSAR criteria. Based on the evaluation criteria addressed in this PC/M, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-072

UNIT : 3 & 4
TURN OVER DATE : 04/17/97

IN-PLACE ABANDONMENT OF LIQUID WASTE DISPOSAL SYSTEM WASTE EVAPORATOR PACKAGE AND SOLID WASTE DISPOSAL SYSTEM HOLDUP AND MIXING

Summary:

This Engineering Package provided the design change documentation required for abandonment of the liquid waste evaporator package and the solid waste disposal system holdup and mixing equipment. The abandoned liquid waste evaporator package included an evaporator, absorption tower, evaporator condenser, vent condenser, distillate cooler, distillate demineralizer, distillate filters, distillate pump, feed preheater assembly, stripping column assembly, and concentrate pump. The abandoned solid waste disposal system holdup and mixing equipment included the radwaste holdup/mixing tanks and their associated pumps. The original purpose of this equipment was to concentrate dissolved and suspended solids from liquid waste using an evaporation, condensation, and filtering process. Over the years, this method of liquid waste disposal has proven too costly to operate and maintain. Presently, all radioactive liquids requiring cleanup are processed by the waste disposal portable demineralizer skid located in the Radwaste Building, and released to the circulating water system upon verification that the discharge meets site release requirements. The only solid wastes currently requiring management and disposal are the spent resins. These are currently transferred to shipping containers, dewatered, and sent off-site for final processing and disposal.

All electrical and mechanical equipment associated with the above liquid and solid waste processing equipment are abandoned in-place. Power feeds were isolated by lifting leads and de-energizing breakers. The various system isolation valves were isolated.

Safety Evaluation:

The liquid waste evaporator package and the solid waste disposal system holdup and mixing tanks and pumps did not serve any safety related functions and were not required to support safe shutdown of the plant. The in-place abandonment of this equipment had no adverse impact on plant safety or plant operations. As demonstrated in the Engineering Package, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-080

UNIT	:	3 & 4
TURN OVER DATE	:	09/18/96

REMOVAL OF LEAD/LAG MODULES FROM LOW PRESSURIZER PRESSURE TRIP INSTRUMENTATION

Summary:

This Engineering Package removed the lead/lag modules from the low pressure portion of the Unit 3 and 4 pressurizer pressure protection channels I, II, and III. The lead/lag units were originally installed in the pressurizer pressure instrumentation loops to provide an amplification of the pressurizer pressure signal as a function of its rate of decline. This was intended to generate an anticipatory low pressure reactor trip signal during a cooldown/depressurization event which otherwise did not warrant a safety injection actuation.

The function of the lead/lag units was not credited in any of the UFSAR safety analyses. The plant safety analyses assume that the reactor trips at the set pressure value which corresponds to the low pressurizer pressure technical specification limit minus the instrument loop uncertainties. Removal of the lead/lag unit does not affect the ability of the pressurizer pressure protection channels to initiate a reactor trip at the assumed setpoint.

Removal of the lead/lag units was based in part on a Westinghouse analysis that concluded lead/lag modules do not perform their intended function. Westinghouse determined through analysis of plant operational transients that events which are severe enough to actuate the low pressurizer pressure trip will generally also actuate the low pressurizer pressure safety injection setpoint, even with lead/lag compensation.

Safety Evaluation:

Removal of the lead/lag units did not affect the ability of the protection channels to trip the reactor due to low pressurizer pressure at the value assumed in the plant safety analyses. In addition, the implemented changes did not affect any setpoints or operating characteristics of any safety system required to prevent or mitigate the consequences of design basis accidents. Since no new failure modes were created by the lead/lag module removal, 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 of the protection channel changes.

PLANT CHANGE/MODIFICATION 95-097

UNIT : 4
TURN OVER DATE : 04/10/96

REPLACEMENT OF MAIN STEAM SAFETY RELIEF VALVE DISCHARGE PIPING

Summary:

This Engineering Package replaced the ten-inch diameter discharge piping for each main steam safety valve (MSSV) on Unit 4 with twelve-inch diameter piping to support plant thermal uprate project implementation. The implementation of this design package involved the replacement of approximately twenty five feet of piping and modifications to two supports for each of the twelve MSSVs. In addition, permanent drains were provided for the discharge piping. The MSSVs serve to protect the secondary plant system from overpressurization. Four MSSVs are located on each of the three main steam lines, upstream of the main steam isolation valves (MSIVs). Based on analyses performed for the plant uprated condition, it was determined that back pressure on the MSSVs resulting from the mass flow rate at uprate conditions coupled with the unusually long discharge lines would be excessive, and could lead to high blowdown rates. High blowdown rates could result in reactor coolant system temperatures dropping below the "no-load" temperature, potentially affecting the fatigue analysis of several reactor coolant system components. In order to preserve all of the reactor coolant system fatigue analysis margins, increasing the diameter of the discharge piping would result in lower backpressures and lower blowdown rates in the range of 3% to 8%, thereby, ensuring that fatigue analysis margins would remain adequate to accommodate reactor coolant system transient conditions.

Safety Evaluation:

The modifications addressed by this Engineering Package ensure that fatigue analysis margins would remain adequate to accommodate reactor coolant system transient conditions. These modifications, therefore, did not alter the design bases, functions, or operation of the main steam safety relief system and did not create any adverse interactions with any other safety related structures or plant systems. Consequently, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-147

UNIT : 3
TURN OVER DATE : 10/03/96

EMERGENCY CONTAINMENT COOLER START LOGIC DESIGN CHANGE

Summary:

This Engineering Package modified the start logic of the emergency containment coolers (ECCs) to support plant thermal uprate project implementation. The original design provided an automatic start capability for all three ECCs. However, analysis has shown that, with the fouling factor assumptions used for thermal power uprate, the component cooling water (CCW) system thermal analysis temperatures could be exceeded if all three ECCs were allowed to start during a limiting case event.

The ECC start logic was changed to coincide with the containment integrity reanalysis performed at uprated conditions. The reanalysis demonstrated that containment temperature and pressure values would remain below design limits provided that, as a minimum, one ECC automatically started at the onset of the event and a second ECC started within 24 hours of the event for containment temperature reduction. The modifications implemented by this Engineering Package revised the ECC start logic such that two ECCs, the dedicated train A and B ECCs, would automatically start on a safety injection signal. The third (i.e., swing) ECC would be capable of manual operation in the event a single active failure caused one of the dedicated train A or B ECCs to fail. Accordingly, this Engineering Package revised the swing ECC control circuit to remove the auto start capability and permit only manual operation.

Safety Evaluation:

The modifications addressed by this Engineering Package and the associated technical specification changes were approved by the NRC in license amendments. Accordingly, the design change to eliminate the automatic start feature of the third (swing) ECC on a safety injection signal had received NRC approval. The implemented changes ensure that CCW system will be capable of performing its safety related heat removal function during a limiting case event. In addition, the capability of the ECCs to perform the required containment pressure mitigation, temperature reduction, and hydrogen mixing functions were not compromised by the design change. Consequently, the ECC control circuit changes did not constitute an unreviewed safety question. Since the emergency containment cooler technical specifications were approved in advance of the Engineering Package implementation, specific NRC approval was not required for implementation of these modifications.

PLANT CHANGE/MODIFICATION 95-148

UNIT : 4
TURN OVER DATE : 10/21/96

EMERGENCY CONTAINMENT COOLER START LOGIC DESIGN CHANGE

Summary:

This Engineering Package modified the start logic of the emergency containment coolers (ECCs) to support plant thermal uprate project implementation. The original design provided an automatic start capability for all three ECCs. However, analysis has shown that, with the fouling factor assumptions used for thermal power uprate, the component cooling water (CCW) system thermal analysis temperatures could be exceeded if all three ECCs were allowed to start during a limiting case event.

The ECC start logic was changed to coincide with the containment integrity reanalysis performed at uprated conditions. The reanalysis demonstrated that containment temperature and pressure values would remain below design limits provided that, as a minimum, one ECC automatically started at the onset of the event and a second ECC started within 24 hours of the event for containment temperature reduction. The modifications implemented by this Engineering Package revised the ECC start logic such that two ECCs, the dedicated train A and B ECCs, would automatically start on a safety injection signal. The third (i.e., swing) ECC would be capable of manual operation in the event a single active failure caused one of the dedicated train A or B ECCs to fail. Accordingly, this Engineering Package revised the swing ECC control circuit to remove the auto start capability and permit only manual operation.

Safety Evaluation:

The modifications addressed by this Engineering Package and the associated technical specification changes were approved by the NRC in license amendments. Accordingly, the design change to eliminate the automatic start feature of the third (swing) ECC on a safety injection signal had received NRC approval. The implemented changes ensure that CCW system will be capable of performing its safety related heat removal function during a limiting case event. In addition, the capability of the ECCs to perform the required containment pressure mitigation, temperature reduction, and hydrogen mixing functions were not compromised by the design change. Consequently, the ECC control circuit changes did not constitute an unreviewed safety question. Since the emergency containment cooler technical specifications were approved in advance of the Engineering Package implementation, specific NRC approval was not required for implementation of these modifications.

PLANT CHANGE/MODIFICATION 95-157

UNIT : 4
TURN OVER DATE : 07/24/96

REMOVE TIME DELAY FOR BLOWDOWN ISOLATION VALVES CV-6275A,B,C

Summary:

This Engineering Package removed the five minute time delay in the opening circuits of steam generator blowdown stop valves CV-4-6275A, B, and C. The time delay was included in the stop valve circuits to allow time for the system bypass valves to open and pressurize the downstream piping between the isolation valves and the blowdown flow control valves (FCV-4-6278A, B, and C), to prevent potential water hammers from occurring. However, due to excessive leakage past the flow control valves, the downstream section of piping could not be pressurized by the bypass flow. Consequently, Operations has modified their operating procedure for the blowdown system such that the bypass valves are not used to pressurize the system, or prevent piping water hammers. The design changes implemented by this Engineering Package were initiated to eliminate an Operator "work-around" item.

The control circuit of valves CV-4-6275A, B, and C was modified to eliminate the opening time delay. In addition, the bypass valve operators were removed to prevent the valves from opening electrically. This maintains the valves in a closed position which is the fail-safe position for operation of the auxiliary feedwater system, containment isolation, and the interlock with sample isolation valves MOV-4-1425, 1426, and 1427. Since position indication for the bypass valves was disconnected with removal of the valve actuator, the upstream manual isolation valve in the bypass loop (SGB-4-082A, B, and C) was locked closed and designated as the containment isolation valve for the bypass portion of the containment penetration. The bypass valves remain as passive elements included in the system seismic boundary.

Safety Evaluation:

The modifications performed on the steam generator blowdown isolation valves by this Engineering Package maintained the existing containment penetration design bases and did not introduce any new failure modes not previously analyzed. The level of protection provided by penetration Nos. 28A, B, and C was enhanced by the new bypass line arrangement because valves SGB-4-082A, B, and C do not have to change state to achieve the isolation function. Consequently, the margin of safety associated with penetrations 28A, B, and C has not decreased as a result of the modifications. 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-168

UNIT : 3
TURN OVER DATE : 06/11/96

REMOVE TIME DELAY FOR BLOWDOWN ISOLATION VALVES CV-6275A,B,C

Summary:

This Engineering Package removed the five minute time delay in the opening circuits of steam generator blowdown stop valves CV-3-6275A, B, and C. The time delay was included in the stop valve circuits to allow time for the system bypass valves to open and pressurize the downstream piping between the isolation valves and the blowdown flow control valves (FCV-3-6278A, B, and C), to prevent potential water hammers from occurring. However, due to excessive leakage past the flow control valves, the downstream section of piping could not be pressurized by the bypass flow. Consequently, Operations has modified their operating procedure for the blowdown system such that the bypass valves are not used to pressurize the system, or prevent piping water hammers. The design changes implemented by this Engineering Package were initiated to eliminate an Operator "work-around" item.

The control circuit of valves CV-3-6275A, B, and C was modified to eliminate the opening time delay. In addition, the bypass valve operators were removed to prevent the valves from opening electrically. This maintains the valves in a closed position which is the fail-safe position for operation of the Auxiliary Feedwater System, containment isolation, and the interlock with sample isolation valves MOV-3-1425, 1426, and 1427. Since position indication for the bypass valves was disconnected with removal of the valve actuator, the upstream manual isolation valve in the bypass loop (SGB-3-082A, B, and C) was locked closed and designated as the containment isolation valve for the bypass portion of the containment penetration. The bypass valves remain as passive elements included in the system seismic boundary.

Safety Evaluation:

The modifications performed on the steam generator blowdown isolation valves by this Engineering Package maintained the existing containment penetration design bases and did not introduce any new failure modes not previously analyzed. The level of protection provided by penetration Nos. 28A, B, and C was enhanced by the new bypass line arrangement because valves SGB-3-082A, B, and C do not have to change state to achieve the isolation function. Consequently, the margin of safety associated with penetrations 28A, B, and C has not decreased as a result of the modifications. 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 of these modifications.

PLANT CHANGE/MODIFICATION 95-170

UNIT : 3
TURN OVER DATE : 01/14/97

THERMAL POWER UPRATE SETPOINT/SCALING

Summary:

The Engineering Package implemented the various instrumentation changes necessary to support the thermal power uprate project. The thermal power uprate affected many of the plant process variables including core Delta-T, decay heat load, steam flow, steam pressure, and feedwater flow. The following parameters and instruments were affected by the uprate: a) Overtemperature Delta-T and Overpower Delta-T reactor trip setpoints, b) rod insertion limit summators, c) Delta-T channel deviation alarm setpoint, d) steam flow / feed flow mismatch comparators and high steam line flow setpoint summators, e) turbine first stage pressure, f) main generator hydrogen temperature alarm setpoint, g) main generator megawatt output, h) spent fuel pool high temperature alarm, i) RCP thermal barrier CCW return high temperature alarm, and j) feedwater pump low suction alarm pressure. FPL received approval from the NRC for the setpoint/scaling changes in advance of implementation of this Engineering Package. The NRC's safety evaluation concluded that operation of the above systems at the proposed power levels was acceptable.

Safety Evaluation:

This RPS and ESFAS setpoint changes addressed by this Engineering Package (including Overtemperature Delta-T, Overpower Delta-T, High Steam Line Flow and Steam Flow / Feed Flow Mismatch) protect the integrity of the fuel and fuel cladding, and mitigate the consequences of design basis accidents. The implemented changes have been shown to be acceptable by reanalyzing all affected accident scenarios. The rescaling of the turbine first stage pressure transmitters further ensures that the high steam line flow setpoint program is enveloped by that used in the revised accident analysis. Turbine first stage pressure also provides input signals to the P-7 logic, which is used to bypass various reactor trip functions below 10% power. The P-7 interlock is now satisfied at approximately 230 MWt rather than 220 MWt. The revised accident analysis demonstrates that this change is also acceptable. The other setpoint/scaling changes within the scope of this Engineering Package relate to control systems and annunciator circuits which are not used to mitigate the consequences of an accident. Consequently, the modifications addressed by this Engineering Package did not constitute an unreviewed safety question. Since the RPS and ESFAS setpoint technical specification changes were approved in advance of the Engineering Package implementation, specific NRC approval was not required for implementation.

PLANT CHANGE/MODIFICATION 95-171

UNIT : 4
TURN OVER DATE : 01/24/97

THERMAL POWER UPRATE SETPOINT/SCALING

Summary:

The Engineering Package implemented the various instrumentation changes necessary to support the thermal power uprate project. The thermal power uprate affected many of the plant process variables including core Delta-T, decay heat load, steam flow, steam pressure, and feedwater flow. The following parameters and instruments were affected by the uprate: a) Overtemperature Delta-T and Overpower Delta-T reactor trip setpoints, b) rod insertion limit summators, c) Delta-T channel deviation alarm setpoint, d) steam flow / feed flow mismatch comparators and high steam line flow setpoint summators, e) turbine first stage pressure, f) main generator hydrogen temperature alarm setpoint, g) main generator megawatt output, h) spent fuel pool high temperature alarm, i) RCP thermal barrier CCW return high temperature alarm, and j) feedwater pump low suction alarm pressure. FPL received approval from the NRC for the setpoint/scaling changes in advance of implementation of this Engineering Package. The NRC's safety evaluation concluded that operation of the above systems at the proposed power levels was acceptable.

Safety Evaluation:

This RPS and ESFAS setpoint changes addressed by this Engineering Package (including Overtemperature Delta-T, Overpower Delta-T, High Steam Line Flow and Steam Flow / Feed Flow Mismatch) protect the integrity of the fuel and fuel cladding, and mitigate the consequences of design basis accidents. The implemented changes have been shown to be acceptable by reanalyzing all affected accident scenarios. The rescaling of the turbine first stage pressure transmitters further ensures that the high steam line flow setpoint program is enveloped by that used in the revised accident analysis. Turbine first stage pressure also provides input signals to the P-7 logic, which is used to bypass various reactor trip functions below 10% power. The P-7 interlock is now satisfied at approximately 230 MWt rather than 220 MWt. The revised accident analysis demonstrates that this change is also acceptable. The other setpoint/scaling changes within the scope of this Engineering Package relate to control systems and annunciator circuits which are not used to mitigate the consequences of an accident. Consequently, the modifications addressed by this Engineering Package did not constitute an unreviewed safety question. Since the RPS and ESFAS setpoint technical specification changes were approved in advance of the Engineering Package implementation, specific NRC approval was not required for implementation.

PLANT CHANGE/MODIFICATION 96-008

UNIT : 3
TURN OVER DATE : 03/25/97

POTENTIAL EDG LOCKOUT FOLLOWING NORMAL STOP

Summary:

This Engineering Package was developed to eliminate an identified "relay race" between the shutdown relay SDRX (energize to run) and the emergency shutdown relay ESDRX (energize to lockout) in the Unit 3 Emergency Diesel Generators' (EDG) control circuitry. The identified relay race could cause an emergency diesel generator to lockout if an emergency start signal was received during the coastdown period from 450 to 40 RPM following a normal stop. To eliminate the potential lockout condition, an interlock was added to the emergency shutdown circuit to ensure pickup of the SDRX relay before pickup of the ESDRX relay during the subject event. The interlock was obtained by adding a new time delay dropout relay in the ESDRX circuit. The addition of this relay does not impact any of the EDG protective trip functions since the only protective trips energized during an emergency start (overspeed and bus differential) initiate an EDG lockout directly, and do not rely on the emergency shutdown circuit. The new seismically qualified, Class E relays were seismically mounted in the 3A and 3B EDG idle start panels. Implementation of the circuit changes was allowed in any plant operating mode, but concurrent outages of the 3A and 3B EDGs were prohibited.

This Engineering Package also corrected several drawing discrepancies that were identified during the EDG circuit review.

Safety Evaluation:

The wiring changes implemented by this Engineering Package enhanced the response of the EDGs to an emergency start signal when received during a normal EDG shutdown. The changes did not impact any other EDG operating scenario. An engineering review demonstrated that the seismic qualification of the panels in which wiring changes were made was not adversely impacted, due to the small weight changes involved. After modifying the emergency shutdown circuit, no new failure modes were identified that could affect EDG and emergency power safety functions. This Engineering Package established requirements for functional testing of the revised circuitry to verify performance. Based on the design package evaluation, these modifications did not have any adverse effects on plant safety or operation. Consequently, these modifications did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the subject modifications.

PLANT CHANGE/MODIFICATION 96-012

UNIT : 3
TURN OVER DATE : 03/24/97

UNIT 3 BORON INJECTION TANK BYPASS MODIFICATION

Summary:

This Engineering Package re-routed the Unit 3 safety injection system piping around the boron injection tank (BIT), due to the potential for stress corrosion cracking in the abandoned tank piping. The BIT was an original plant design feature that was rendered obsolete by a re-analysis of the main steam line break (MSLB) event performed for the replacement steam generators in the early 1980s. Over the past fifteen years, the BIT and associated piping has remained a passive part of the safety injection system pressure boundary. Based on an engineering review of observed stress corrosion cracks in the chemical and volume control system boric acid supply piping, the BIT safety injection piping was identified as potentially subject to the same stress corrosion cracking failure mechanisms. Removing the BIT from the pressurized flow path of the safety injection system and re-routing the system piping was determined to be the best technical and most cost effective option. Accordingly, this design package provided a new simplified safety injection flow path, which accommodated the in-situ abandonment of the Unit 3 BIT and associated piping and equipment. Implementation of the piping changes was allowed in plant operating Modes 5, 6, or defueled. Specific system configuration restrictions were imposed by the Engineering Package to ensure that the four high head safety injection pumps remained operable to support Unit 4 operation, in accordance with plant technical specification requirements.

Safety Evaluation:

The modifications addressed by this Engineering Package were shown to have a negligible affect on the hydraulic performance of the safety injection system during the injection and recirculation phases of an accident, including refueling water storage tank (RWST) drain down time. The piping configuration changes did not alter the response of the Safety Injection System to postulated accident conditions. In addition, no new failure modes were introduced by the flow path change. Based on the design bases and installation criteria evaluated 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 of the piping system changes.

PLANT CHANGE/MODIFICATION 96-022

UNIT	:	3 & 4
TURN OVER DATE	:	01/29/97

THERMAL POWER UPRATE IMPLEMENTATION

Summary:

This Engineering Package evaluated the power escalation process for thermal uprate, defined specific requirements for escalation above 2200 MW_t, and provided changes to the various engineering documents affected by the change in plant operating parameters, including the UFSAR, Design Basis Documents (DBDs), and Vendor Technical Manuals (VTMs). No hardware changes or accident analysis revisions were specifically implemented by this document. It essentially provided an overview of the impact of the thermal uprate project including associated hardware changes, accident analysis revisions, licensing basis document changes, recommended Emergency Operating Procedure changes, and recommended Off-Normal Operating Procedure changes.

FPL received approval from the NRC for the safety related and power generation system hardware changes and accident reanalyses in advance of implementation of this Engineering Package. The NRC's safety evaluation concluded that operation of Turkey Point Units 3 and 4 at the proposed power levels was acceptable.

Safety Evaluation:

Implementation of the thermal uprate project did not involve any unusual operating practices or maintenance practices. No changes were made to the designs or methods of operation of components postulated to initiate design basis events such that the probability of their failure was increased. The restrictions and precautions imposed on the power escalation process assured that the plant operating configuration remained within accident analysis limits. The plant procedures associated with power escalation were revised to accommodate uprate for normal, off-normal, and emergency conditions, and were consistent with pre-uprate operating practices and methodology. Since no new equipment or operating practices were invoked for thermal uprate, the modifications did not constitute an unreviewed safety question. The associated technical specification changes for thermal uprate were approved in advance of the Engineering Package implementation, thus, specific NRC approval was not required for implementation the proposed changes.

UNIT : 4
TURN OVER DATE : 05/22/96

SPRING/SETPOINT CHANGE FOR THE PILOT OPERATED LOCKUP VALVES FOR THE ECC CCW SUPPLY/RETURN ISOLATION VALVES

Summary:

This Engineering Package installed stiffer springs in the pneumatic actuators of the component cooling water (CCW) supply and return isolation valves for the emergency containment coolers (ECCs), to enhance the fail-open operation of the valves on loss of instrument air. The actuators utilize a spring-assisted pilot operated lock-up valve (POLV) to provide the fail safe operation when instrument air pressure decreases below a nominal 45 psig value. Failure of the lock-up valves to slide to the fail safe position on loss of instrument air has been observed during periodic surveillance tests. In each case, an increased friction force on the POLV O-rings prevented the valves from failing open. To overcome the increased drag force on the POLV spool, a replacement spring with a significantly higher spring rate was installed. The air pressure setpoint at which the POLV actuates was also increased from 45 psig to 60 psig.

This Engineering Package also made the POLV test valves installed by Temporary System Alteration (TSA) 4-96-30-07 a permanently installed part of the CCW return valve actuators. These test valves allow the POLV to be tested without physically stroking the isolation valve; thereby improving valve reliability. Permanent installation of the test valves does not impact the function of the POLV, the actuator, or the control valve.

Safety Evaluation:

The valve actuator changes implemented by this Engineering Package enhanced the response of the ECCs to accident conditions, when instrument air is not available. The changes did not impact the ECC control circuit or actuation logic. An engineering review demonstrated that the seismic qualification of the supply and return valves was not adversely affected by the modification due to the small weight changes involved. The replacement springs were shown to be consistent with the original spring material, and compatible with the POLV materials of construction. In addition, functional testing of the modified actuators verified proper performance of the new POLV spring and setpoint change. Based on the design package evaluation, these modifications did not have any adverse effects on plant safety or operation. Consequently, these modifications did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the subject modifications.

PLANT CHANGE/MODIFICATION 96-039

UNIT : 3
TURN OVER DATE : 05/22/96

SPRING/SETPOINT CHANGE FOR THE PILOT OPERATED LOCKUP VALVES FOR THE ECC CCW SUPPLY/RETURN ISOLATION VALVES

Summary:

This Engineering Package installed stiffer springs in the pneumatic actuators of the component cooling water (CCW) supply and return isolation valves for the emergency containment coolers (ECCs), to enhance the fail-open operation of the valves on loss of instrument air. The actuators utilize a spring-assisted pilot operated lock-up valve (POLV) to provide the fail safe operation when instrument air pressure decreases below a nominal 45 psig value. Failure of the lock-up valves to slide to the fail safe position on loss of instrument air has been observed during periodic surveillance tests. In each case, an increased friction force on the POLV O-rings prevented the valves from failing open. To overcome the increased drag force on the POLV spool, a replacement spring with a significantly higher spring rate was installed. The air pressure setpoint at which the POLV actuates was also increased from 45 psig to 60 psig.

This Engineering Package also made the POLV test valves installed by Temporary System Alteration (TSA) 3-96-30-06 a permanently installed part of the CCW return valve actuators. These test valves allow the POLV to be tested without physically stroking the isolation valve; thereby improving valve reliability. Permanent installation of the test valves does not impact the function of the POLV, the actuator, or the control valve.

Safety Evaluation:

The valve actuator changes implemented by this Engineering Package enhanced the response of the ECCs to accident conditions, when instrument air is not available. The changes did not impact the ECC control circuit or actuation logic. An engineering review demonstrated that the seismic qualification of the supply and return valves was not adversely affected by the modifications due to the small weight changes involved. The replacement springs were shown to be consistent with the original spring material, and compatible with the POLV materials of construction. In addition, functional testing of the modified actuators verified proper performance of the new POLV spring and setpoint change. Based on the design package evaluation, these modifications did not have any adverse effects on plant safety or operation. Consequently, these modifications did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the subject modifications.

PLANT CHANGE/MODIFICATION 96-040

UNIT : 3 & 4
TURN OVER DATE : 02/13/97

APPENDIX R DOCUMENTATION CHANGE IN SUPPORT OF CRs 96-023, 96-474, 96-754, 96-951, 96-1060, & 96-1152

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. 96-023, 96-474, 96-754, 96-951, 96-1060, and 96-1152.

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.

PLANT CHANGE/MODIFICATION 96-076

UNIT : 3
TURN OVER DATE : 03/17/97

ENHANCEMENT OF THE RCP BUS UNDERVOLTAGE TIME DELAY RELAY CIRCUIT

Summary:

This Engineering Package replaced the reactor coolant pump (RCP) bus undervoltage time delay relay in each reactor protection system (RPS) train (UVTD-A & UVTD-B) with two parallel undervoltage time delay relays. The reactor coolant system low flow protection subsystem is part of the reactor protection system (RPS) which provides reactor trips to protect the core from exceeding departure from nucleate boiling (DNB) limits during a loss of reactor coolant flow. The RCS low flow protection subsystem includes trips for low flow measured in the primary piping, RCP bus undervoltage, RCP breaker position, and RCP bus underfrequency. A reactor trip for RCP bus undervoltage is initiated by a one out of two coincident logic of the undervoltage relays from both 4160 volt buses. After the one of two coincidence logic is satisfied, the UVTD-A and UVTD-B relays provide a 1.0 second time delay before the reactor trip (RT) relays drop out. However, the existing time delay relays are electrically downstream of this coincidence logic. As a result, a single failure of any one time delay relay has the potential to initiate a reactor trip. The new relays will be installed in parallel and will provide coincidence logic, so that, a two out of two failure logic is required for an erroneous reactor trip. This will preclude the possibility of a single relay failure initiating a reactor trip, thus improving plant reliability for power generation. A single failure that prevents the relay contacts from changing state on coil deenergization (e.g., contacts stick or become welded) would not adversely affect operation of the RPS due to the availability of the redundant train circuitry.

Safety Evaluation:

These modifications were implemented during a refueling shutdown in accordance with requirements imposed by plant technical specifications. This design modification did not alter the logic of any safety function of the RPS and will enhance the reliability of the RPS, protecting it from erroneous reactor trips originating from the single failure of an RCP bus undervoltage timed delay relay. This Engineering Package did not alter any design bases, safety analysis, operational or testing requirements of the reactor protection system. Consequently, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 96-084

UNIT	:	3
TURN OVER DATE	:	03/28/97

RADIANT ENERGY SHIELDS INSIDE CONTAINMENT

Summary:

This Engineering Package added stainless steel sheet metal lagging over existing Thermo-Lag wrapped raceways inside containment to satisfy 10CFR50, Appendix R, Section III.G.2.f requirements for radiant energy shields. Stainless steel sheet metal is known for its ability to provide radiant energy shielding and has been used in the past inside containment to protect various components and cables. The addition of stainless steel lagging on these components augmented the existing Thermo-Lag material. The lagging provided: a) an augmented radiant energy shield for the Thermo-Lag and protected raceway for a minimum of 30 minutes, b) a vapor barrier that prevents open flaming from developing on the surface of the Thermo-Lag (if subjected to fire), and c) a moisture shield that prevents the Thermo-Lag from falling away from the installed location if subjected to fire and/or hose streams.

The lagging was installed snugly around the Thermo-Lag covered components to prevent moisture intrusion. Stainless steel banding and/or stainless steel pop rivets were used to anchor the lagging in place.

Safety Evaluation:

The modifications performed by this Engineering Package restored the 30 minute fire rating to protected raceways inside containment, and resolved the combustibility and degradation issues surrounding Thermo-Lag radiant energy shields. An engineering review demonstrated that the seismic qualification of the modified conduits and terminal/pull boxes would not be adversely impacted by the addition of lagging materials, due to the small weight changes involved. The review also demonstrated that the lagging would remain in place during a design basis seismic event, and not become a source of debris for the containment sump screens. The review concluded that the lagged raceways would be enveloped by the existing ampacity correction factors, and that none of the affected circuits would have to be derated. Since the installation of lagging on affected raceway components did not change the operation, function, or design basis of any structure, system, or component important to safety, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 96-085

UNIT : 4
TURN OVER DATE : 10/02/97

RADIANT ENERGY SHIELDS INSIDE CONTAINMENT

Summary:

This Engineering Package added stainless steel sheet metal lagging over existing Thermo-Lag wrapped raceways inside containment to satisfy 10CFR50, Appendix R, Section III.G.2.f requirements for radiant energy shields. Stainless steel sheet metal is known for its ability to provide radiant energy shielding and has been used in the past inside containment to protect various components and cables. The addition of stainless steel lagging on these components augmented the existing Thermo-Lag material. The lagging provided: a) an augmented radiant energy shield for the Thermo-Lag and protected raceway for a minimum of 30 minutes, b) a vapor barrier that prevents open flaming from developing on the surface of the Thermo-Lag (if subjected to fire), and c) a moisture shield that prevents the Thermo-Lag from falling away from the installed location if subjected to fire and/or hose streams.

The lagging was installed snugly around the Thermo-Lag covered components to prevent moisture intrusion. Stainless steel banding and/or stainless steel pop rivets were used to anchor the lagging in place.

Safety Evaluation:

The modifications performed by this Engineering Package restored the 30 minute fire rating to protected raceways inside containment, and resolved the combustibility and degradation issues surrounding Thermo-Lag radiant energy shields. An engineering review demonstrated that the seismic qualification of the modified conduits and terminal/pull boxes would not be adversely impacted by the addition of lagging materials, due to the small weight changes involved. The review also demonstrated that the lagging would remain in place during a design basis seismic event, and not become a source of debris for the containment sump screens. The review concluded that the lagged raceways would be enveloped by the existing ampacity correction factors, and that none of the affected circuits would have to be derated. Since the installation of lagging on affected raceway components did not change the operation, function, or design basis of any structure, system, or component important to safety, these 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 of these modifications.

PLANT CHANGE/MODIFICATION 96-093

UNIT : 4
TURN OVER DATE : 10/08/97

UNIT 4 ADDITION OF CCW HEAD TANK

Summary:

This Engineering Package installed a static head tank above the existing component cooling water (CCW) system surge tank to increase the operating pressure of the system, and resolve potential voiding concerns identified in Generic Letter 96-06. Functions such as insurge capacity, normal operating band, system vent, etc. were transferred from the surge tank to the new head tank. The head tank was sized smaller than the existing surge tank but the existing water volume and operating range was retained by the modified design. A variety of miscellaneous changes were implemented to accommodate both the increase in normal system operating pressure (static head increase) and the reduced insurge capacity. These included instrument changes, relief valve setpoint changes, logic change for RCV-4-609, increase in the excess letdown heat exchanger design rating, and replacement of the MOV-4-716A&B spring packs to accommodate the required increase in closing torque.

No physical modifications were necessary to adapt the existing CCW surge tank to water solid operation. A low level alarm, however, was added to the surge tank to assist Operations personnel with level control during periods of reduced CCW inventory.

Safety Evaluation:

The modifications addressed by this Engineering Package eliminated a potential failure mode for the CCW system and restored its capability to operate as a subcooled system in accordance with the original design intent. The increase in CCW system normal and transient operating pressures were analyzed and found to be within applicable design code requirements. In addition, all of the components installed by this Engineering Package were designed to accommodate the applicable UFSAR seismic and hurricane wind loads. The logic change for RCV-4-609 did not adversely affect any system safety function. Modifications performed on the excess letdown heat exchanger and system relief valves ensured that system inventory and function would not be adversely affected by anticipated overpressure events. Based on the evaluation criteria contained in the Engineering Package, these 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 of the CCW system changes.

PLANT CHANGE/MODIFICATION 97-012

UNIT : 4
TURN OVER DATE : 10/10/97

THERMAL OVERPRESSURIZATION OF ISOLATED PIPING

Summary:

This Engineering Package modified several sections of water-filled piping inside containment to eliminate the potential for isolated segments to become thermally overpressurized from heating during accident conditions. This potential failure mode was identified in Generic Letter 96-06 as a condition that could jeopardize the ability of systems to perform their safety related functions, and potentially lead to a loss of containment integrity.

Those piping segments potentially susceptible to thermally induced overpressurization were modified by either a) installing a thermal relief valve within the isolated boundary, b) modifying the failure position of an affected boundary valve from fail-close to fail-open, or c) drilling a 1/8" diameter hole in an affected boundary valve disc.

The new thermal relief valves were generally set to relieve at pressures 10% higher than the lowest rated component within the isolated bounds. The relief valves were installed on the inside containment portion of affected piping associated with containment penetrations. The discharge of each relief valve was piped to a collection tank inside containment.

Safety Evaluation:

The modifications addressed by this Engineering Package eliminated a potential failure mode for various water-filled piping systems inside containment. The provision of thermal relief paths 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 UFSAR to ensure that the installation did not create the possibility of any adverse interactions with safety related structures, systems and components. Those modifications made to system boundary valves maintained the existing containment penetration design bases and did not introduce any new failure modes not previously analyzed. 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 97-026

UNIT : 4
TURN OVER DATE : 10/06/97

MOV-4-744A AND MOV-4-744B REPACKING AND EQUALIZING LINE INSTALLATION

Summary:

This Engineering Package installed bonnet equalizing lines on the residual heat removal (RHR) system discharge isolation valves, MOV-4-744A and MOV-4-744B. These equalizing lines allow any potential fluid trapped in the bonnet cavities to be vented back to the reactor coolant system (RCS) during depressurization events, and eliminates the potential for high pressure fluid remaining in the bonnets to hydraulically lock the valves in the closed position.

Installation of the equalizing lines required that the existing packing leak-off lines be cut, capped, and abandoned in place. The lower set of packing rings also had to be removed from the valve stuffing boxes to establish the necessary vent path between the bonnet and the packing leakoff port. The equalizing lines converge downstream of the valves and form a common vent path to the RCS. Connection to the RCS is at an existing vent valve location. The installation required approximately 10 feet of 3/8" stainless steel tubing, associated tube clamps, and supports.

A check valve was installed in each equalizing branch line and a single manual isolation valve was installed in the common tubing run. The manual isolation valve is required to be locked in the open position during plant operation.

Safety Evaluation:

The modifications addressed by this Engineering Package eliminated a potential failure mode for the RHR system. No new failure modes were created by the modified stuffing box arrangement or the equalizing line installation. The new tubing and supports were evaluated in accordance with seismic criteria contained in the UFSAR to ensure that the installation did not create the possibility of any adverse interactions with safety related structures, systems and components. The components installed by this Engineering Package (i.e., 10 feet of stainless steel tubing, associated tube clamps, supports, and valves) were found to have a negligible impact on the existing emergency core cooling system (ECCS) heat sink analysis, and hydrogen generation analysis. 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 97-035

UNIT	:	3 & 4
TURN OVER DATE	:	08/27/97

MONITOR TANK "B" DIAPHRAGM ELIMINATION

Summary:

This Engineering Package removed the diaphragm in the "B" boric acid recycle monitor tank, commonly known as the monitor tank, because it was torn and no longer needed to satisfy a system functional requirement. The primary function of the tank diaphragm was to maintain the discharge from the plant evaporators in a degassed state so that it could be returned to the primary water storage tank (PWST) without further processing. The tear causes the diaphragm to settle below the water level blocking the outlet nozzle, and impeding normal tank drain down efforts.

Presently, the monitor tank is used for processing laundry waste water. Since laundry waste water is not returned to the PWST, preventing oxygenation of the tank contents is no longer a plant design requirement.

The diaphragm was removed from the tank by trimming the membrane material as close to the interior tank attachment bracket as possible.

Safety Evaluation:

The Monitor Tank does not serve any safety related functions and is not required to support safe shutdown of the plant. Removal of the tank diaphragm was evaluated and found to have no adverse affect on plant safety or plant operations. In addition, the removal process did not impact the existing tank structure or original design code. As demonstrated in the Engineering Package, 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 design change.

PLANT CHANGE/MODIFICATION 97-041

UNIT : 4
TURN OVER DATE : 10/11/97

CORE EXIT THERMOCOUPLE COLUMN REMOVAL - UNIT 4

Summary:

This Engineering Package permanently removed the upper portion of core exit thermocouple column 53 from the reactor vessel because it was damaged beyond repair during removal of the reactor vessel closure head, in preparation for the Unit 4 Cycle 17 refueling outage. The damaged instrument column was removed by cutting the column support tube and enclosed conduits at an elevated location above the upper support plate and sealing the lower portion of the instrument column (and exposed conduit ends) with weld metal. The reactor vessel head penetration was sealed using the existing seal configuration with a specially fabricated core exit thermocouple nozzle plug that duplicated the existing thermocouple column seal block. The abandoned core exit thermocouples were part of the inadequate core cooling system (ICCS) instrumentation, which was designed to yield information on emergency core cooling system operation, and long term surveillance of post-accident decay heat removal. During normal plant operation, it was possible to use the core exit temperature information to confirm reactor core design parameters and calculate the quadrant power tilt ratio. The core exit thermocouples are classified as Regulatory Guide 1.97 Type A Category 1 variables. Abandonment of the core exit thermocouples did not impact the control room qualified safety parameter display system (QSPDS).

Safety Evaluation:

The modifications described in this Engineering Package permanently disabled 13 core exit thermocouples and left a portion of the intact support column attached to the reactor vessel internals. Due to the passive nature of the thermocouple instruments and the thermocouple column sealing device, no new failure modes were created by the design change. The configuration of the reactor vessel upper internals package and the head nozzle plug assembly were analyzed and determined to meet all applicable load combinations. Since the number of remaining operable core exit thermocouples continued to satisfy technical specification requirements, and FPL's commitment to provide core exit temperature monitoring during mid-loop operations, there was no safety significance associated with the abandonment of the damaged core exit thermocouple column. Based on the evaluation criteria contained in the Engineering Package, the design change did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for implementation.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION JPN-PTN-SEEJ-88-042
REVISION 8

UNIT : 4
APPROVAL DATE : 09/05/97

DE-ENERGIZATION OF UNIT 4 4160 VOLT SAFETY RELATED BUSSES

Summary:

This evaluation was developed to establish the requirements and restrictions which must be placed on the operation of Units 3 and 4 and their equipment when a Unit 4 4160 volt bus was de-energized and train "A" and "B" load centers were 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 4 4160 volt bus de-energization. The de-energization of a Unit 4 4160 volt safety related bus, with Unit 4 in cold or refueling shutdown (Modes 5 and 6) or de-fueled and Unit 3 at power operation (Mode 1) or below, is sometimes necessary to allow for 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 8 updated the electrical design configuration and imposed new restrictions to accommodate modifications being implemented by PC/M 96-096, "C-Bus Reliability Improvements Modifications."

Safety Evaluation:

This safety evaluation addressed the technical and licensing requirements for the de-energization of each Unit 4 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 UFSAR or the results and conclusions of any previous safety evaluations. The actions or procedural changes identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. 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 actions or precautions identified in this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEEJ-89-085
REVISIONS 10, 11, and 12

UNIT	:	3
APPROVAL DATES	:	Rev.10 02/20/97
		Rev.11 03/09/97
		Rev.12 03/13/97

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 was de-energized and Train "A" and "B" load centers were 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.

Revisions 10, 11, and 12 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 UFSAR 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 plant changes in 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.

UNIT : 3
APPROVAL DATE : 03/26/97

10 CFR 50.59 SAFETY EVALUATION - ACCEPTABILITY OF AS-FOUND CONDITION FOR RHR CHECK VALVE 3-753A

Summary:

This safety evaluation examined the as-found metallurgical defects in the residual heat removal (RHR) system 3A pump discharge check valve 3-753A. In response to Significant Operating Experience (SOER) 86-3, Turkey Point implemented a disassembly and inspection program on a sampling basis to ensure check valve internals were intact and were not experiencing abnormal wear. During visual inspection of the 3A RHR pump discharge check valve, three linear indications were identified on the valve seat. One of the indications cut across the stellite seat and extended into the austenitic stainless steel valve body. A liquid penetrant examination determined that the other two defects met the acceptance criteria of ASME Section III. A flaw evaluation was conducted consistent with the analytical flaw evaluation methods contained in ASME Section XI (IWB-3600). Based on this review and the material behavior for the cast austenitic stainless steel valve body, the only relevant degradation expected was fatigue. Due to the low calculated crack growth for the estimated valve duty cycles, it was concluded that the valve would provide acceptable operation until the end-of-service life of the plant.

Revision 6 documented the valve re-inspection performed during the Unit 3 Cycle 16 refueling outage. The inspection revealed that there was no increase in the length of the flaw on the valve seat.

Safety Evaluation:

Re-inspection of the linear indications showed that no crack growth occurred over the approximate 60 month operating period. This confirmed the stationary nature of the linear indications and the conservatism of previous analyses. Since the as-found condition of the valve will not impact the capability of the RHR system to perform its safety functions (effectively until the end-of-service life of the plant), the actions or 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 conditions identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-93-010
REVISION 2

UNIT : 4
APPROVAL DATE : 09/04/97

SAFETY EVALUATION FOR ICW VALVE REPLACEMENT AND B HEADER CRAWL THROUGH INSPECTION

Summary:

This safety evaluation addressed nine intake cooling water (ICW) isolation valves which were scheduled to be replaced during the 1993 Unit 4 refueling outage. A crawl through inspection and repair of the Unit 4B ICW header and the C ICW pump discharge piping was performed during the same outage. The purpose of this safety evaluation was to assess all potential safety concerns caused by activities associated with valve replacement and the piping crawl through inspection/repair. Some of the valve replacements required each of the ICW headers to be individually removed from service. To ensure residual heat removal (RHR) and spent fuel pit (SFP) cooling requirements were met, ICW operations were controlled in accordance with the applicable technical specifications and system operating procedures.

Revision 2 addressed replacement of valve 4-50-310 during the 1997 refueling outage, and the crawl through inspection and repair of the B ICW header.

Safety Evaluation:

Because plant operating procedures assure that decay heat removal through the RHR system and the SFP cooling system will be maintained, the ICW valve replacement and the inspection/repair crawl through did not adversely impact plant safety and operation. Removal of valve bodies and piping components during the replacement and crawl through activities were evaluated and did not affect the seismic qualifications of the remaining piping or its operability. The piping met the allowable stresses as defined in the UFSAR. The actions or plant changes (procedures and/or hardware) 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 JPN-PTN-SECS-93-013
REVISION 2

UNIT : 3 & 4
APPROVAL DATE : 09/04/97

SAFETY EVALUATION FOR SCAFFOLDING AND COVER FOR SPENT FUEL POOL ROOMS

Summary:

This evaluation addressed the installation of scaffolding and pool covers in the Units 3 and 4 spent fuel pool rooms to allow the application of coatings to walls and ceiling areas. The purpose of this safety evaluation was to demonstrate that the proposed installation of temporary scaffolding, pool covers, and the use of the spent fuel pool cranes during coating application would not adversely affect plant safety or operation during any mode, except while the affected unit is in Mode 6, or while handling fuel or inserts in the affected spent fuel pool. These restrictions limited pool heat load to a recently discharged 1/3 core, plus past discharged cores, consistent with the design basis of the spent fuel cooling system. Scaffolding was installed to provide access to all or most walls, and ceiling areas not located directly above the pools. The spent fuel pool gantry crane was used as a traveling staging platform to access adjacent walls, ceiling areas located directly above the pools, and potentially for installation of the pool cover system.

Revision 2 included calculated pool heatup data supporting the initial 1/3 core restriction. It also included a Deviation Request Form to allow Engineering to review minor deviations from the cover installation details.

Safety Evaluation:

The temporary structures covered by the scope of this safety evaluation were designed to withstand all applicable loads, including seismic loads. These temporary structures did not modify or actively interact with any plant equipment important to safety. No new permanent plant equipment was added by the activity, and restrictions were placed in this evaluation to preclude implementation of the coating work in Mode 6, or while fuel or inserts were handled in the affected pool. The actions or plant 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.

<u>SAFETY EVALUATION JPN-PTN-SENP-95-007</u>
REVISION 2

UNIT : 3
APPROVAL DATE : 09/10/96

SAFETY EVALUATION FOR OPERABILITY OF RHR DURING INTEGRATED SAFEGUARDS TESTING

Summary:

This safety evaluation addressed engineered safeguards integrated testing (ESIT), performed during each refueling outage, with respect to a generic Westinghouse concern related to the decay heat removal capabilities of the steam generators (SGs) during plant shutdowns in Mode 5. Westinghouse identified that during Mode 5 shutdowns, there was the potential for gas formation within the steam generator U-tubes, which makes the use of steam generators and natural circulation for decay heat cooling ineffective. Thus, in accordance with plant technical specifications, both trains of the residual heat removal system (RHR) must remain operable during the period of safeguards testing performed in Mode 5. Since safeguards testing was normally performed during Mode 5 and could have involved the potential for temporary alterations in engineered safety features systems (including the RHR system), this evaluation was developed to document that the RHR system remained operable during the period when safeguards testing was conducted with the plant in Mode 5. 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 address the generic Westinghouse concerns, since the ESIT procedures ensure that both RHR trains remain operable at all times during the ESIT without having to perform any compensatory actions.

Revision 2 addresses the temporary test configuration of each 4160 volt bus during the ESIT. The evaluation concluded that both busses will remain available to support RHR pump operability.

Safety Evaluation:

The engineered safeguards integrated test (ESIT) is performed each refueling outage to demonstrate the readiness of emergency power systems and safeguards equipment. This evaluation examined the operability of both RHR trains during ESIT testing because of the potential for steam generator U-tube voiding in Mode 5. This evaluation concluded that ESIT procedures ensure that both RHR trains remain operable at all times during testing without requiring compensatory actions. Consequently, ESIT procedures did not involve an unreviewed safety question nor did they require changes to plant technical specifications. Therefore, prior NRC approval was not required to perform the next outage related ESIT safeguards tests or to address the generic implications of SG voiding as this could have applied to existing ESIT procedures.

SAFETY EVALUATION JPN-PTN-SENP-95-023
REVISIONS 2 and 3

UNIT	:		4
APPROVAL DATES	:	Rev.2	10/01/97
		Rev.3	10/01/97

SAFETY EVALUATION FOR OPERABILITY OF RHR DURING INTEGRATED SAFEGUARDS TESTING

Summary:

This safety evaluation addressed engineered safeguards integrated testing (ESIT), performed during each refueling outage, with respect to a generic Westinghouse concern related to the decay heat removal capabilities of the steam generators (SGs) during plant shutdowns in Mode 5. Westinghouse identified that during Mode 5 shutdowns, there was the potential for gas formation within the steam generator U-tubes, which makes the use of steam generators and natural circulation for decay heat cooling ineffective. Thus, in accordance with plant technical specifications, both trains of the residual heat removal system (RHR) must remain operable during the period of safeguards testing performed in Mode 5. Since safeguards testing was normally performed during Mode 5 and could have involved the potential for temporary alterations in engineered safety features systems (including RHR), this evaluation was developed to document that the RHR system remained operable during the period when safeguards testing was conducted with the plant in Mode 5. 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 address the generic Westinghouse concerns.

Revision 2 addresses performance of the ESIT with Unit 4 in Mode 6 with reactor coolant system level two feet below the reactor vessel flange. Revision 3 addresses RHR operability during test transients with respect to technical specification compliance. It is concluded that both 4160 volt busses will remain available to support RHR pump operability.

Safety Evaluation:

This safety evaluation examined the operability of both RHR trains during ESIT testing because of the potential for steam generator U-tube voiding in Mode 5. The examination concluded that the ESIT procedures ensure that both RHR trains remain operable at all times during testing without requiring compensatory actions. Consequently, ESIT procedures did not involve an unreviewed safety question nor did they require changes to plant technical specifications. Therefore, prior NRC approval was not required to perform the next outage related ESIT safeguards tests or to address the generic implications of SG voiding as this could have applied to existing ESIT procedures.

UNIT : 3 & 4
APPROVAL DATE : 09/11/96

SAFETY EVALUATION FOR UNIT 3 AND UNIT 4 TWENTY-FIFTH YEAR CONTAINMENT TENDON SURVEILLANCES

Summary:

Containment tendon surveillance testing is performed every fifth year from the date of the initial containment structural integrity test, in accordance with plant technical specifications. This safety evaluation examined the implementation activities (e.g., mobilization, equipment erection on the containment dome, heavy load lifts, extraordinary weather precautions, demobilization activities) associated with the twenty-fifth year tendon surveillances on Units 3 and 4 to ensure that adverse interactions with safety related equipment are precluded during the heavy load handling and "construction-like" activities associated with tendon testing. Appendix B to this safety evaluation contained a list of the restrictions imposed on the overall test sequence. The safety evaluation concluded that sufficient precautions and limitations exist to permit performance of the surveillance tests in any plant operating mode.

Safety Evaluation:

This safety evaluation examined the construction and testing equipment to be used during the surveillance test and demonstrated that any potential adverse interactions could be accommodated by a) the design attributes of the surveillance equipment and structures, b) the facilities with which the surveillance equipment or structures could interact, or c) the restrictions imposed by Appendix B of the safety evaluation. In addition, the safety evaluation examined the detensioning procedure and concluded that the surveillance activity will not affect the ability of the containment to withstand maximum design basis loads. The safety evaluation concluded that the activities associated with twenty-fifth year tendon surveillance test 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 JPN-PTN-SENS-95-052
REVISIONS 1 and 2

UNIT	:		3
APPROVAL DATES	:	Rev.1	03/03/97
		Rev.2	03/13/97

SAFETY EVALUATION FOR REVISION OF UNIT 3 DIESEL FUEL OIL TRANSFER SYSTEM TECHNICAL SPECIFICATION BASES

Summary:

An earlier evaluation, JPN-PTN-SENS-95-050, documented the operability of the Unit 3 emergency diesel generators (EDGs) and compliance with technical specification requirements for EDG diesel fuel oil transfer from the storage tank to the day tank using manual actions. This safety evaluation addressed changes in the UFSAR to clarify the use of manual actions to fulfill the Unit 3 EDG fuel transfer function on loss of instrument air, and proposed changes in the bases of technical specifications to clarify the design differences between Unit 3 and Unit 4 diesel fuel transfer systems. The diesel fuel transfer valves on Unit 3 are designed to fail closed on loss of instrument air which prevents the automatic transfer of fuel oil from the Unit 3 diesel oil storage tank to the EDG day tanks. The use of portable compressed air bottles to manually restore function to the EDG automatic fill isolation valves were proceduralized as part of the controlling procedure for loss of instrument air. This evaluation concluded that the use of manual actions to fulfill the Unit 3 EDG fuel transfer function is within the current design basis of the plant.

Revision 1 clarified the position that the use of manual actions in place of automatic operation is an acceptable compensatory measure under loss of instrument air conditions, and is consistent with the design intent. Revision 2 further clarified that the EDGs are operable under loss of instrument air conditions, provided the fuel transfer valves can be operated with a portable air source.

Safety Evaluation:

This evaluation supports revisions to technical specification bases for the Unit 3 EDG diesel fuel transfer system surveillance requirements which reflect that design basis EDG diesel fuel transfer functions are satisfied by using either existing proceduralized manual actions on loss of instrument air or by automatic action of the fill transfer isolation valves. Since manual actions to fulfill the Unit 3 EDG fuel oil transfer functions are within the current plant design basis, this evaluation concluded that the proposed clarifications to technical specification bases and the UFSAR 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 JPN-PTN-SEMP-96-004
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 02/27/97

EVALUATION OF AP ARMAFLEX INSULATION FOR CONDENSATION DURING TEMPORARY CONTAINMENT COOLING

Summary:

This safety evaluation addressed the Appendix R requirements for the use of foam insulation which was installed on portions of the component cooling water (CCW) system. Under a separate Plant Change/Modification (PC/M) 95-054, permanent plant modifications were implemented to allow the use of chilled water in the CCW system to provide chilled water to the normal containment coolers (NCCs) to cool the containment buildings during refueling outages. The PC/M identified the permanent installation of foam insulation for portions of the affected CCW piping to minimize condensation within the auxiliary building. The areas identified were the Unit 3 and 4 boric acid evaporator rooms, the Unit 3 pipe and valve room and other areas as required. This safety evaluation addressed the Appendix R requirements for the use of the foam insulation and associated material qualifications and suitability requirements. Based on a review of the material properties, FPL determined that the material was acceptable for use in the auxiliary building. The evaluation within PC/M 95-054 addressed the piping and pipe supports associated with temporary containment cooling.

Revision 1 revised the scope of the safety evaluation to include the additional CCW piping that was insulated under PC/M 95-177.

Safety Evaluation:

This safety evaluation concluded that the use of insulating foam on the CCW piping did not adversely affect the performance of CCW pumps or heat exchangers, since the basis for system heat removal did not consider the piping as a heat transfer surface. The evaluation also examined the seismic performance, the additional mass attached to CCW system piping, and the chemical composition, flame retardant characteristics, and temperature rating of the foam insulation. The total amount of foam material added to each fire zone was evaluated. The use of the foam insulation did not adversely affect CCW system performance or plant operations. Consequently, the use of foam insulation on selective portions of the CCW piping did not constitute an unreviewed safety question or require changes to the plant technical specifications. Based on the above, prior NRC approval was not required for the use of foam insulation on those portions of CCW piping identified within this safety evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-96-014
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 02/06/97

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 (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 is 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.

Safety Evaluation:

This evaluation addressed the use of ultrafine filter cartridges for the RCS, seal water return, and seal water injection 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 JPN-PTN-SENP-96-017
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 05/09/96

SAFETY EVALUATION FOR UFSAR CONSISTENCY REVIEW AND UPDATE

Summary:

In response to NRC Information Notice 95-54, a self-assessment activity was undertaken to review the Turkey Point UFSAR and determine the nature and extent of discrepancies between UFSAR descriptions and the design and procedural configuration of the plant. Chapters 3, 4, 6, 7, 8, 9, 11, 12, and 14 of the UFSAR were included in the review. Ninety-six findings were identified by the self-assessment team and forwarded to Engineering for review, final disposition, and incorporation into the UFSAR as appropriate. Further review by engineering found that 67 of the findings were already addressed by in-process change documents, or determined not to be a discrepancy. Twenty-four of the findings identified administrative clarifications or addressed historical information in the UFSAR. Five of the 96 findings documented statements in the UFSAR that were inconsistent with the plant design or plant operating procedures. No operability issues were identified and no physical design changes were required to resolve the discrepancies. Attachment 1 to this safety evaluation lists each review finding, its classification, and status. Attachments 2 through 6 contain the FSAR User Comment Forms for the five identified discrepancies. Attachment 7 documents the resolution of each finding.

Safety Evaluation:

The safety evaluation documented that all UFSAR findings were dispositioned and that none of the discrepancies impacted plant safety or operation. The evaluated discrepancies did not change the operation, function, or design bases of any structure, system, or component important to safety as described in the UFSAR. Consequently, the UFSAR 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 JPN-PTN-SENS-96-024
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 03/21/97

SAFETY EVALUATION TO SUPPORT A CROSS-CONNECTED RHR SYSTEM REALIGNMENT DURING A REFUELING OUTAGE

Summary:

This safety evaluation was developed to address the impact of a proposed residual heat removal (RHR) system realignment on plant technical specifications during refueling activities. This temporary realignment was required to ensure that RHR cooling would remain available to remove decay heat during the period when repairs were performed on the nozzle of the 4B RHR heat exchanger. At the same time these repairs were performed, the 4B RHR pump was unavailable due to an outage on the 4B 4160 volt bus. Additionally, maintenance was performed on the RHR pump discharge check valves, and IST surveillance testing was required to ensure operability of these valves. This evaluation assessed the sequence of RHR pump discharge check valve IST surveillance testing to ensure the credited RHR "loop" remained operable. As a result of these refueling outage maintenance activities, the operable RHR loop required by Technical Specifications was composed of the 4A RHR pump combined with the 4B RHR heat exchanger. Technical Specifications require one operable and operating RHR loop while in operating Mode 6, provided that the refueling pool water level remains 23 feet or more above the reactor vessel flange. In addition to RHR technical specification requirements, procedures imposed stringent administrative controls on support systems associated with each operable RHR decay heat cooling loop. These procedures address Component Cooling Water (CCW) and Intake Cooling Water (ICW) systems and their power supplies.

Revision 1 extended the scope of the evaluation to include the generic alignment of any RHR pump with its opposite train heat exchanger.

Safety Evaluation:

This evaluation examined critical design and licensing criteria which involved single failure criteria, support system alignments, power supply alignments, and seismic criteria. This evaluation found that the temporary realignment of RHR and associated maintenance activities did not adversely affect safety functions and met all technical specification and procedural requirements. Consequently, the RHR realignment and associated maintenance activities did not involve an unreviewed safety question or require changes to plant technical specifications. Therefore, prior NRC approval was not required for implementation of the RHR realignment and other actions identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SENP-96-025
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 06/04/96

SAFETY EVALUATION FOR REMOVAL OF THE RCC CHANGE FIXTURE FROM THE FSAR

Summary:

This safety evaluation was prepared to delete the description of the rod control cluster (RCC) change fixture from the UFSAR. The RCC change fixture is a necessary refueling device during fuel reloads performed with an in-core fuel shuffle. This was the practice at Turkey Point until the replacement of the steam generators in the early 1980's. Subsequently, refuelings have been performed by offloading the entire core to the spent fuel pool and performing the RCC change activities in the spent fuel pool. The RCC change tool used in the spent fuel pool provides an equivalent function to the RCC change fixture inside containment. The proposed UFSAR changes were provided as Attachment 1 to the safety evaluation.

Safety Evaluation:

Abandoning the RCC change fixture in place and removing reference to it in both the UFSAR and plant procedures did not alter the design basis, functions, or operation of any safety related equipment. The practice of performing full core offloads has been evaluated and found to improve safety with respect to a number of concerns, including reduced consequences for refueling cavity seal failures and minimizing operating time at reduced inventory conditions. The actions and 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 JPN-PTN-SENS-96-028
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 09/19/96

SAFETY EVALUATION UFSAR OPERATIONAL REVIEW AND UPDATE

Summary:

In response to NRC Inspection Report Nos. 50/250-94-01 and 50/251-94-01, a self-assessment activity was undertaken to review the Turkey Point UFSAR and determine the nature and extent of discrepancies between UFSAR descriptions and the plant procedures. The review consisted of selected portions of the FSAR identified as containing operational or procedural information. Forty-five findings were identified by the self-assessment team and forwarded to Engineering for review, final disposition, and incorporation into the UFSAR as appropriate. Further review by engineering found that 21 of the findings were already addressed by in-process change documents, or determined not to be a discrepancy. Sixteen of the findings identified administrative clarifications or addressed historical information in the UFSAR. Eight of the 45 findings documented statements in the UFSAR that were inconsistent with the way in which the plant was operated. No operability issues were identified and no physical design changes were required to resolve the discrepancies. Attachment 1 to this safety evaluation lists each review finding, its classification, and status. Attachments 2 through 12 contain the FSAR User Comment Forms for the identified discrepancies.

Safety Evaluation:

The safety evaluation documented that all UFSAR findings were dispositioned and that none of the discrepancies impacted plant safety or operation. The evaluated discrepancies did not change the operation, function, or design bases of any structure, system, or component important to safety as described in the UFSAR. Consequently, the UFSAR 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 JPN-PTN-SECS-96-033
REVISIONS 1 and 2

UNIT	:		4
APPROVAL DATES	:	Rev.1	08/29/96
		Rev.2	04/25/97

SAFETY EVALUATION FOR INSTALLATION OF REMOTE CAMERA FOR 'B' RCP OIL LEVEL VERIFICATION

Summary:

This safety evaluation addressed the installation of a remote camera assembly inside containment to monitor the 4B reactor coolant pump (RCP) motor oil level. The assembly consisted of a bent steel conduit with a miniature video camera and light attached to the bent end. The total weight of the conduit and camera assembly was approximately 10 pounds, resulting in insignificant loads being applied to the RCP motor oil piping and shield wall. The video feed from the camera was routed to a communication box near the elevator platform on the 30'-6" elevation of the containment building. The video cable was connected to spare telephone leads which terminated outside containment in the cable spreading room.

Revision 1 addressed minor administrative changes to the evaluation. Revision 2 addressed relocation of the camera assembly to monitor leakage at the pump seal housing to main flange joint.

Safety Evaluation:

An engineering review demonstrated that the seismic qualification of the RCP motor oil piping and supports would not be adversely impacted the addition of the camera and conduit assembly, due to the small weight changes involved. The review also demonstrated that the camera assembly would remain in place during a design basis seismic event, and not damage adjacent equipment considered important to safety. Since the installation of camera assembly 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 JPN-PTN-SEMS-96-037
REVISION 0

UNIT : 4
APPROVAL DATE : 05/31/96

SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF A FREEZE SEAL AND BLIND FLANGE PER TSA-04-96-046-11 TO SUPPORT MOV-4-350 VALVE REPAIRS

Summary:

This safety evaluation addressed the use of a freeze seal to temporarily isolate a section of the chemical and volume control system (CVCS) to perform maintenance activities on the emergency boration valve, MOV-4-350. The freeze seal was to be applied downstream of check valves 4-351 and 4-352 as a backup feature in the event that check valve 4-351 leaked past its seat. The freeze seal was considered to provide a housekeeping function since the upstream check valves provided the pressure boundary function for the charging pump suction header during the maintenance evolution. This evaluation addressed plant operation with the emergency boration line isolated during the repair process, and the potential impact on charging system operation due to freeze seal failures. 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. A blind flange was installed at the upstream flange connection for FT-4-110 to allow the boric acid flow path from the Boric Acid Storage Tanks (BASTs) to be maintained as an operable flowpath during the maintenance evolution.

Safety Evaluation:

This evaluation addressed the temporary uncoupling of the emergency boration flowpath caused by removal of FT-4-110, the impact on plant operation, and the various precautions imposed to ensure the safe conduct of maintenance. Strict controls were imposed on the freeze seal process and contingency measures were developed to establish pressure boundary integrity for the open system should the freeze seal start to thaw. Based on the precautions identified, the evaluation concluded that the maintenance could be performed, and that this 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 activities identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-96-038
REVISIONS 0 and 1

UNIT	:		3
APPROVAL DATES	:	Rev.0	08/22/96
		Rev.1	03/31/97

SAFETY EVALUATION FOR UNIT 3 STEAM GENERATORS' SECONDARY SIDE FOREIGN OBJECTS

Summary:

This evaluation addressed the potential safety significance of operating the Unit 3 steam generators (SG) with foreign objects present in the secondary side. The foreign objects identified within the scope of this evaluation are those which are considered to be irretrievable. Previously, individual safety evaluations have addressed the acceptability of continued Unit 3 operation while these foreign objects remained in the steam generators and associated systems. The purpose of this evaluation was: (1) to re-examine the analyses, results, requirements, and restrictions of previous evaluations while applying recent industry standards; (2) to document the methodology for determining the interval between steam generator eddy current tests as affected by estimated steam generator tube wall wear times; and (3) to provide a single Unit 3 safety evaluation to assess and document all the Unit 3 steam generator foreign object estimated wear times as adjusted by updated steam generator eddy current data and steam generator Foreign Object Search and Retrievals (FOSAR) results.

Revision 1 addressed the impact of one new unretrievable foreign object identified in the 3B steam generator during the Cycle 16 refueling outage.

Safety Evaluation:

Previous safety evaluations prepared for each SG secondary side foreign object have considered the effects of the object upon tube integrity, chemistry, SG instrumentation, the main steam system, and SG blowdown and sampling systems. This evaluation established current wear time to minimum tube wall thickness estimates based on conservative assumptions from Westinghouse WCAP-14258 and associated Westinghouse clarification correspondence. These wear times assume worst case conditions and actual wear times are likely to be much greater than the Westinghouse methodology would predict. Based on this assessment, this evaluation determined that currently identified foreign objects within the secondary side of the Unit 3 steam generators did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for continued operation of the plant with foreign objects present in the secondary side of the steam generators, or endorsement of the programmatic actions identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-96-040
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 07/25/96

SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF DRAIN HOSES AND PERFORMANCE OF HOT SPOT FLUSHES ON THE RHR SYSTEM PER TP-96-046 AND TP-96-047

Summary:

This safety evaluation examined the procedure for flushing the residual heat removal (RHR) system with water from the refueling water storage tank, to eliminate radioactive hot spots at various system drain locations. The affected drains were located in the suction piping from the refueling water storage tank and south containment recirculation sump, and at the RHR heat exchangers. The flushes were performed by installing a flush adapter and tygon hose to the discharge of each drain valve, routing the hose to the nearest suitable floor drain, and opening the drain valve for approximately 20 seconds. The affected drain valves were flushed one at a time while the RHR system remained operable in the normal standby valve lineup (Modes 1 - 3). A flushing flow rate of 45 gpm was expected through the drain piping based on the static head of the refueling water storage tank. As a precautionary measure, an additional ball valve was used with the flush adapter on the refueling water storage tank suction piping drains, to provide a backup isolation capability (in lieu of RWST isolation) in the event that the piping drain valve could not be re-closed when the flushing activity was complete.

Safety Evaluation:

This evaluation addressed the temporary configuration of the system with the installed flushing adaptor, the impact on plant operation, and the various precautions imposed to ensure safe conduct of the maintenance activity. Strict controls were imposed on the flushing process and contingency measures were developed to establish pressure boundary integrity for the open system should a drain valve fail to re-close, or actuation of the engineered safety features occur. Based on the precautions identified, the evaluation concluded that the maintenance activity could be performed in Modes 1 - 3, and that this 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 activities identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEIS-96-042
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/23/96

SAFETY EVALUATION FOR THE OPERATION OF RAD-*-6458 THE REACTOR VESSEL HEAD LEAK DETECTION SYSTEM ON A NON-CONTINUOUS BASIS

Summary:

This evaluation analyzed the impact associated with operation of the reactor vessel head leak detection system on a non-continuous basis, rather than a continuous basis as described in the UFSAR. The reactor vessel head leak detection system was installed at the request of plant management in the late 1980's to enhance reactor vessel head leak detection capability. The system is designed to detect an increasing trend in radioactivity level in the control rod drive mechanism (CRDM) cooler ductwork over containment background levels which would indicate the presence of a reactor vessel head leak. Performance of the system is based on the capability of the skid to detect a difference in activity levels between the two sample points. Due to the recent upgrade in containment atmosphere gaseous and particulate radiation monitoring capability, efforts to more thoroughly seal the reactor coolant system (RCS) pressure boundary (e.g., CRDM canopy seal clamps, crush resistant O-rings), and a lessened tolerance for RCS leakage, the usefulness of the reactor vessel head leak detection system has diminished since its original installation. Rather than removing the system completely, this safety evaluation provided the necessary justification to reduce its hours of operation, and reduce the dose rate received by maintenance personnel attempting to maintain the system operating in a continuous sampling mode.

Safety Evaluation:

The reactor vessel head leak detection system was installed at the request of plant management and did not perform a safety related function. Changing its operating philosophy did not adversely affect any structure, system, or component considered important to safety. Since no physical plant changes are required to implement the new monitoring scheme, 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 or changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEIS-96-044
REVISION 0

UNIT	:	3 & 4
APPROVAL DATE	:	08/01/96

SAFETY EVALUATION FOR ABANDONING SPENT FUEL PIT BRIDGE CRANE LASER RANGE FINDER

Summary:

This safety evaluation analyzed the impact on plant safety and operation associated with abandonment of the laser range finding systems on the Unit 3 and 4 spent fuel pit bridge cranes. The existing equipment was considered to be broken beyond repair and new units, different in form and fit, would be required to restore the systems to operable status. Use of the fixed scale and pointer system is preferred by plant personnel for fuel handling since it is simple, accurate, reliable, and maintenance-free. Additionally, the scale and pointer is system is referenced directly to individual fuel storage cells, whereas the laser range finding system indicates the distance in feet from the spent fuel pit walls, and does not provide direct indication of storage rack positions. The laser range finding system is physically and electrically independent of the crane's load handling system. The required UFSAR changes were provided as Attachment 1 to the safety evaluation.

Safety Evaluation:

The laser range finding system did not serve any safety related functions and had no effect on the operational capability of the spent fuel pit bridge crane, or its load handling system. Since the in-place abandonment of this equipment had no adverse impact on plant safety or fuel handling operations, it was concluded that the actions and 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-SEES-96-046
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/26/96

SAFETY EVALUATION FOR ESSENTIAL EQUIPMENT AFFECTED BY A HEAVY LOAD DROP RESULTING FROM EPS MODIFICATIONS

Summary:

This safety evaluation re-verified the acceptability of the safe load paths for the turbine gantry crane and spent fuel pool cask crane in light of discrepancies noted between the plant procedure for heavy load handling, and commitments made to the NRC regarding NUREG 0612. In addition, field walkdowns identified the presence of several conduits installed during recent plant modifications such as the Emergency Power Upgrade Project which had the potential to be impacted by a heavy load drop. The areas of the plant examined in the safety evaluation included a) the area north of the Unit 3 switchgear rooms, b) the auxiliary building roof and volume control tank roof areas, and c) the areas above the Unit 3 and 4 component cooling water room roof grating. Consistent with NUREG 0612, this evaluation used the basic criterion that sufficient equipment be available to bring the plant to cold shutdown conditions during a postulated heavy load drop event. The essential equipment required for safe shutdown was based on the established Appendix R Essential Equipment List. The Appendix R Safe Shutdown Analysis was used as guidance in developing the applicable safe shutdown assumptions.

Safety Evaluation:

The safety evaluation demonstrated that additional restrictions were required to preserve safe shutdown capability during heavy load handling activities north of the Unit 3 switchgear rooms. Consequently, additional restrictions were imposed on turbine gantry crane operation to preclude heavy load handling in the affected area. These actions 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 changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEMS-96-048
REVISION 0

UNIT : 4
APPROVAL DATE : 08/27/96

SAFETY EVALUATION FOR MEASURING ICW HEADER FLOW PER TP-96-070

Summary:

This safety evaluation was prepared to evaluate a temporary re-alignment of the intake cooling water (ICW) system for flow measurement purposes in accordance with Temporary Procedure TP 96-70. Obtaining an accurate measurement of ICW flow was a commitment made in LER 50/250-95-003, "Intake Cooling Water System Flow Rate Found less Than Required By Design Basis," to facilitate an assessment of the condition of the ICW basket strainers. The proposed ICW system alignment was similar to that used for ICW pump in-service testing in that two ICW pumps were aligned to the operable header supplying two component cooling water (CCW) heat exchangers, with the third ICW pump aligned to the two turbine plant cooling water (TPCW) heat exchangers and the third CCW heat exchanger. The CCW heat exchanger on the inoperable header was required to be taken out of service during the test with the CCW flow isolated. The test was performed by closing the basket strainer outlet isolation valves to establish zero flow for calibration and then incrementally throttling the valves open to obtain the required flow data. The test alignment required entry into the 72 hour technical specification Action Statement for one inoperable ICW header. In addition, the test alignment caused increased flows through the TPCW heat exchangers. The TPCW heat exchangers are located downstream of the safety related (Quality Group C) boundary and serve non-safety related functions only.

Safety Evaluation:

The ICW and CCW systems are capable of performing their design basis heat removal functions with the minimum amount of equipment required to be operable by plant technical specifications. Although a relaxation of the single failure criterion is permitted when operating under a technical specification Action Statement, contingency measures were developed to restore the normal ICW and CCW system alignments should an emergency or abnormal ICW temperatures occur. The actions authorized 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 test procedure.

SAFETY EVALUATION PTN-ENG-SECS-96-059
REVISION 0

UNIT : 3
APPROVAL DATE : 09/25/96

EVALUATION OF TEMPORARY SCAFFOLDING WITHIN CONTAINMENT FOR USE IN REPLACING RELIEF VALVE RV-3-203

Summary:

This safety evaluation addressed the installation of scaffolding inside containment to allow replacement of the Unit 3 letdown line relief valve, RV-3-203. The purpose of the evaluation was to demonstrate that the temporary scaffolding structure would not adversely affect plant safety or operation if erected in Mode 3 or below. The evaluation examined the potential for adverse seismic interactions between the scaffolding and adjacent safety related equipment, the potential impact on containment free volume and heat sink analyses due to the metal scaffold support members, the potential impact on the containment hydrogen generation analysis due to the zinc based scaffold support member coating, the potential interaction with the containment sump, and containment combustible loading.

Safety Evaluation:

The temporary structure covered by the scope of this safety evaluation was designed to withstand all applicable loads, including seismic loads. The temporary structure did not modify or actively interact with any plant equipment important to safety. An engineering review demonstrated that the applicable containment analyses were not affected by the composition of the scaffolding structure (i.e., zinc, metal, wood) due to the small amount of material involved. The actions or plant 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-96-062
REVISION 1

UNIT : 3
APPROVAL DATE : 09/27/96

SAFETY EVALUATION TO PERMIT WIDER PRT LEVEL OPERATING BAND

Summary:

This safety evaluation was prepared to address plant operation with the pressurizer relief tank (PRT) level indicator LT-3-470 operating beyond its stated accuracy. Vendor literature indicated that the nominal instrument error should be 1% (indicated error) while actual in-situ operation displayed an error of approximately 3%. The purpose of this safety evaluation was to address operation of the plant with the existing operating level band assuming a higher instrument error. The setpoints associated with PRT operation were established by Westinghouse. The low level setpoint of 68% (± 1% accuracy) is intended to insure that the design basis pressurizer steam space release would not result in exceeding the PRT design limit of 200 °F. The 83% high level limit is specified such that on a pressurizer steam space release, the pressure in the PRT would not exceed 50 psig. Indication of PRT level is a Regulatory Guide 1.97, Category D variable. The stated regulatory guide function is to monitor plant operation.

Revision 1 addressed the impact of the higher instrument error on the reactor coolant system leak rate calculation.

Safety Evaluation:

The PRT does not perform a safety related function. Consequently, inadvertent operation of the PRT rupture disk due to level uncertainty is not a malfunction of equipment that could adversely affect the plant's ability to respond to an accident or transient. The 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 changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SECS-96-065
REVISIONS 0 and 1

UNIT	:		3
APPROVAL DATES	:	Rev.0	12/17/96
		Rev.1	01/30/97

10 CFR 50.59 SAFETY EVALUATION FOR UNIT 3 CYCLE 16 REFUELING OUTAGE TURBINE OVERHAUL SUPPORT

Summary:

This safety evaluation was prepared for the Unit 3 Cycle 16 refueling outage to address a) use of the Turkey Point Units 1 & 2 turbine gantry crane on the Unit 3 turbine operating deck; b) lifting and transport of several heavy turbine components over safety related equipment and outside the approved safe load paths; and c) transport of the high pressure turbine rotor from the Unit 4 laydown area to the turbine staging area located south of the site cafeteria building.

Several restrictions were identified in the evaluation to permit use of the Units 1 & 2 turbine gantry crane near safety related equipment. In addition, compensatory measures applicable to both cranes were identified to address lifting and transport of the turbine loads outside of the safe load path zones. Rigging options were also identified as defense-in-depth protection from load drops. A review of plant drawings and procedures was performed to evaluate the heavy haul route from the laydown area to the turbine staging area. The review concluded that the only structure of concern for the proposed heavy haul route was the underground intake cooling water (ICW) pipes, located just south of the Unit 4 laydown area. This section of piping was previously evaluated for the haul route of a main station transformer and a turbine-generator rotor assembly. It was concluded that the movement of the HP rotor assembly was encompassed by the two previous haul route evaluations, and that the existing utilities along the proposed route would not be adversely affected by the subject activity.

Revision 1 addressed comments from the Plant Nuclear Safety Committee and other administrative items.

Safety Evaluation:

The safety evaluation addressed the various failure modes and effects of handling heavy loads over safety related equipment. Strict administrative controls and compensatory measures were imposed to ensure that safety related equipment under the crane travel path would not be adversely affected by a heavy load drop. 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 perform the planned turbine overhaul activities.

SAFETY EVALUATION PTN-ENG-SENS-96-068

REVISION 0

UNIT	:	3 & 4
APPROVAL DATE	:	12/05/96

SAFETY EVALUATION RELATED TO REPOSITIONING OF VALVE HV-7 ON THE POST ACCIDENT CONTAINMENT VENTILATION SYSTEM

Summary:

This safety evaluation was prepared to assess the acceptability of maintaining valve HV-7 on the post accident containment ventilation (PACV) system normally closed during plant operation. Maintaining this valve in a normally closed position reduces the potential to overpressurize the PACV filters and allows greater operational flexibility for the post accident hydrogen monitoring (PAHM) system and post accident sampling system (PASS). The potential to overpressurize the PACV filters with the current valve alignment was documented in a condition report and involved cross connecting the "A" and "B" PASS/PAHM system sensing lines in an alternate sample alignment. To demonstrate the acceptability of leaving valve HV-7 in a normally closed position, a detailed dose assessment of operator action to re-open the valve was performed based on the time when operation of the PACV system is predicted (17 days post-accident).

Safety Evaluation:

Operation with valve HV-7 closed eliminates a potential failure mode for the PACV filter housings. The evaluation demonstrated that operator action to re-open the valve for PACV operation would not result in acceptable dose consequences. Since no new failure modes were created by the change in valve position, it was concluded that the actions and 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 for implementation of the actions or changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEMS-96-078
REVISION 0

UNIT : 3
APPROVAL DATE : 12/09/96

SAFETY EVALUATION RELATED TO TSA No. 03-96-20-09 FOR REPAIR OF MOV-3-832

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with a proposed temporary alteration of the primary water makeup and component cooling water (CCW) systems to permit repair of the MOV-3-832 valve internals. Valve MOV-3-832 isolates the primary water makeup line to the component cooling water (CCW) system and provides the code boundary break between the Quality Group C (CCW) and Quality Group D (primary makeup water) piping. This evaluation addressed a) the use of a blind flange to temporarily maintain the primary water makeup system pressure boundary, and b) the temporary relocation of the Quality Group C code boundary to valves 3-711A (or 3-737C if 3-711A is closed) and 3-711B. The evaluation addressed the design and material qualifications of the blind bonnet flange for MOV-3-382. It also addressed the impact on plant operation associated with local manual operation of valves 3-711A (or 3-737C) and 3-711B for CCW makeup, versus control room operation of MOV-3-382. The evaluation concluded that the use of a field operator to initiate CCW makeup was acceptable given the maximum permissible leak rate for the system and the normal inventory stored in the surge tank. To satisfy the safety related isolation function, the evaluation required that a dedicated operator be stationed at the valves during the makeup process so that they could be quickly closed if required by the control room.

Safety Evaluation:

This evaluation addressed the temporary plant conditions and restrictions imposed on operation of the CCW system while valve MOV-3-382 was out of service. The evaluation also addressed issues, such as, seismic effects, failure modes and effects, technical specification requirements, isolation boundaries, and CCW makeup capabilities. The proposed actions did not adversely affect any safety related functions. The evaluation concluded that the maintenance could be performed, and that the temporary system alterations 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 activities identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEES-96-081
REVISION 0

UNIT : 4
APPROVAL DATE : 12/13/96

SAFETY EVALUATION FOR TEMPORARY SYSTEM ALTERATION (TSA) NO. 4-96-003-16 FOR CONTAINMENT INSTRUMENT AIR BLEED VALVE CV-4-2826 CIRCUIT MODIFICATION

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with a temporary modification of the containment instrument air bleed valve control circuit, to restore valve operability. The containment instrument air bleed valve (CV-4-2826) is part of the containment isolation feature for penetration No. 63 and is normally open during plant operation to prevent containment pressurization due to the accumulation of instrument air exhausting (i.e., bleeding) from pneumatically operated components inside the building. The temporary circuit modification was required to circumvent a faulty power cable in the CV-4-2826 solenoid valve control circuit. The modification consisted of abandoning the faulty cable in place and utilizing spare conductors in an adjacent cable associated with CV-4-2826 to provide the power feed. Since valve Cv-4-2826 is located in the pipe and valve room, all of the circuit modifications were performed in the auxiliary building. The evaluation considered the affects of postulated high energy line breaks, cable separation, and post-modification testing.

Safety Evaluation:

This evaluation examined the potential for new failure modes. It was concluded that the circuit changes did not alter the operation of the valve, its actuation logic, or interlocks. Design basis issues such as cable separation and the environmental qualification of cable splices were also addressed. The actions or plant changes in procedures, design documents, and/or hardware 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-97-006
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 02/11/97

SAFETY EVALUATION TO ABANDON AUXILIARY STEAM IN THE AUXILIARY AND RADWASTE BUILDINGS

Summary:

The Turkey Point auxiliary steam system was included as part of the original plant design to supply low pressure saturated steam to the auxiliary feedwater pump turbines, boric acid evaporators, waste disposal evaporators, boric acid batching tank, gas strippers, and control building unit heating. By the mid-1980s, much of this equipment was no longer being used because the boric acid recycle and waste disposal processes proved too costly to operate and maintain. Engineering Packages were prepared to address the in-place abandonment of the boric acid recycle and waste disposal equipment, including isolation of the auxiliary steam system valves (see PC/Ms 94-141 and 95-072 in Section I). This safety evaluation provides the justification to further abandon the auxiliary steam system desuperheater stations, condensate recovery transfer pumps, and remaining auxiliary steam components inside the auxiliary building and radwaste building. It was prepared to support Minor Engineering Package 95-081 which provides the implementing instructions necessary to mechanically and electrically isolate the above equipment. It primarily addresses the abandonment of auxiliary steam to the boric acid batching tank and the ability to satisfactorily mix sodium tetraborate decahydrate (i.e., Borax) at temperatures below 55 °F for post-LOCA chemical injection. It concludes that the technical specification requirement to maintain the boric acid storage tank room temperature above 55 °F does not apply to Borax batching in the batching tank and provided a basis to reduce the minimum water temperature for batching post-LOCA chemicals to 39 °F.

The supply of auxiliary steam to the AFW pump turbines was not affected by the proposed changes.

Safety Evaluation:

The safety evaluation demonstrated that the auxiliary steam system did not perform any safety related functions and was not required to support safe shutdown of the plant. The in-place abandonment of this equipment had no adverse impact on plant safety or plant operations. Consequently, the actions or plant changes in procedures, design documents, and/or hardware 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-SEIS-97-013
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 02/27/97

SAFETY EVALUATION FOR THE ABANDONMENT, REPAIR, OR RESTORATION OF RVLMS SENSORS

Summary:

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-IC-007 in the reactor vessel level monitoring system (RVLMS) sensor maintenance process, in lieu of the current practice, which requires that a Plant Change Modification (PC/M) package be issued and implemented to evaluate the work scope and provide any necessary drawing revisions. It also demonstrated that the specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-IC-007 was developed to streamline the repair and abandonment process for defective sensors outside of the PC/M process. It allows Maintenance to abandon or repair defective sensors as well as restore presently abandoned or modified sensors upon installation of new RVLMS probes using the "Specification Clarification" process. Sensor abandonment is permitted if the technical specification requirement of four or more operational sensors per channel exist before and after the activity. Sensor repair is permitted if the technical specification requirement will be met following the repair activity. The only repair activity permitted by the specification involves a failed unheated thermocouple. In these cases, repair is accomplished by jumpering the failed unheated thermocouple to the nearest operable unheated thermocouple. This repair process is justified on the basis that there is not a substantial temperature difference between adjacent sensors. The specification also permits the wiring restoration of a given channel upon installation of a new RVLMS probe.

Safety Evaluation:

The RVLMS sensors provide a means for acquiring information about the core, but do not perform any type of control function. Since the number of remaining operable sensors continues to satisfy technical specification requirements, there was no safety significance associated with the abandonment, repair, or restoration process described in the generic specification. The provisions of the generic specification 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 engineering provisions of the generic specification identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEMS-97-014
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 05/29/97

SAFETY EVALUATION TO ADDRESS THE OPERATION OF THE CONDENSATE POLISHING DEMINERALIZER SYSTEM

Summary:

This safety evaluation was prepared to update the UFSAR so that it accurately reflects the current operational requirements of the condensate polishing demineralizer system. The condensate polishing demineralizer system was originally designed to purify condensate from the condenser hotwell by filtration and demineralization to provide high quality feedwater to the steam generators. Operation and control of the system is independent from the existing condensate and feedwater system. The system was taken out of service to address a unit reliability issue. On two previous occasions, during unit startup, while manipulating valves to place the system in service, a valve malfunction caused condensate to be diverted to the canal which resulted in a steam generator feedwater pump trip (low suction pressure), and the subsequent automatic initiation of auxiliary feedwater. As a result of these events, FPL has decided that the condensate polishing demineralizer system will only be used prior to unit startups. The current FSAR description indicated that the system was still a normal part of the feedwater flow path during unit operation.

Revision 1 addressed PNSC comments related to the 50.59 safety evaluation and proposed FSAR changes.

Safety Evaluation:

The effects of using the condensate polishing demineralizer system during unit startup, and isolating the system during plant operation, was evaluated and determined not to have an adverse effect on plant safety or operation. The evaluation examined the potential impact on steam generator secondary side degradation, auxiliary feedwater system performance, and offsite doses due to steam generator tube rupture events. The proposed change in operating philosophy and resulting UFSAR 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 JPN-PTN-SEMS-97-016
REVISION 0

UNIT : 3
APPROVAL DATE : 03/18/97

SAFETY EVALUATION FOR INSTALLATION OF AN OIL DRAIN LINE ON THE 3B RCP MOTOR

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with the temporary installation of an oil drain line on the 3B reactor coolant pump (RCP) motor. The drain line was being installed to mitigate the effects of an oil leak at the flywheel seal on the upper reservoir. According to Westinghouse, the leak was most likely caused by air becoming entrained in the oil. It was speculated that the entrained air was being released under the flywheel seal area and forcing an oil mist or foam up through the seal. The new drain line was intended to provide an additional vent to the upper reservoir to relieve any pressure buildup under the flywheel seal. It would also direct any leaking oil to the oil collection system, away from any hot reactor coolant system piping. The drain line installation required 12 feet of 3/8-inch diameter stainless steel tubing.

Safety Evaluation:

The temporary drain line was designed to withstand all applicable loads, including seismic loads. The installation did not modify or actively interact with any plant equipment important to safety. An engineering review demonstrated that the applicable containment analyses were not affected by the materials of construction (i.e., metal) due to the small amount of material involved. The actions or plant 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-SEIS-97-017
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 05/20/97

SAFETY EVALUATION RELATED TO THE ABANDONMENT OF MONITORING FUNCTION OF PARTICULATE DETECTORS OF THE SPING EFFLUENT MONITORS

Summary:

This safety evaluation assessed the acceptability of abandoning the on-line particulate monitoring function of the Special Particulate, Iodine and Noble Gas (SPING) monitors due to the high number of failures associated with Channel 1 (beta particulate detector) and Channel 2 (alpha particulate detector) of the steam jet air ejector (SJAE) units. The SPING monitoring system was installed to satisfy the post-TMI regulatory requirements (NUREG 0737 and Regulatory Guide 1.97) for high range radioactive gas monitoring and iodine and particulate monitoring. In addition, the system provides a backup to the process radiation monitoring system and provides plant release data for Technical Specification Section 6.8. Plant operating history has shown that the SJAE SPING channel failures were primarily due to the effects of secondary system chemistry in the sample stream. The Channel 1 & 2 detectors are located immediately behind the particulate filter on the sample skid and are exposed to high concentrations of ammonia. These high concentrations of ammonia cause corrosion and damage the mylar window of the scintillation detector. The SPING vendor, Eberline, does not currently make a detector that can survive the SJAE ammonia environment.

In accordance with plant technical specifications and the Off Site Dose Calculation Manual (ODCM), the required SPING channels are the noble gas channels (Channels 5, 7, and 9, and Channel 10 for the plant vent SPING). There are no requirements to maintain the particulate monitoring capabilities of Channels 1 & 2. Particulate monitoring is accomplished by collecting samples on filter media and analyzing them in the onsite chemistry laboratory.

Safety Evaluation:

The SPING monitors do not perform a safety related function. It was demonstrated that abandonment of Channels 1 & 2 would not adversely affect plant operation, the plant technical specifications, ODCM, or any regulatory commitments made pursuant to NUREG 0737 or Regulatory Guide 1.97. Consequently, the actions or plant changes in procedures, design documents, and/or hardware 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-97-023
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 06/13/97

SAFETY EVALUATION RELATED TO SEVERE ACCIDENT MANAGEMENT GUIDELINE IMPLEMENTATION

Summary:

This safety evaluation was prepared to document FPL Engineering's review of the Turkey Point Severe Accident Guidelines to demonstrate consistency with the Westinghouse Owner's Group Severe Accident Management Guidance (SAMG) document prepared for member utilities. The need to address plant response to a severe accident, and develop appropriate strategies for dealing with these "beyond design basis" events, was first addressed by the NRC in a commission position paper and later linked to the resolution of NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." In addition to the development of Probabalistic Safety Assessments for individual plants, the NRC desired the development of accident management strategies for severe accidents. In response to this aspect of the severe accident issue, the Westinghouse Owner's Group developed the SAMG for use at member utilities. FPL subsequently used the SAMG to develop a set of Severe Accident Guidelines (SAGs) that are specific to Turkey Point. Transition steps from the existing Emergency Operating Procedures (EOPs) to the new SAGs were also developed and included in the Engineering review. The Engineering review concluded that the SAGs were consistent with the Westinghouse guidance document and that no changes to the UFSAR or DBDs were required. The evaluation identified several additional EOPs that were required to have transition points to the new SAGs.

Safety Evaluation:

Overall implementation of the SAGs was evaluated against the criteria of 10 CFR 50.59 due to its interfaces with the EOPs. The proposed EOP changes did not alter any of the strategies for coping with design basis events. The EOP transition points to the SAGs only occur after all required emergency actions have been completed and are unsuccessful. Consequently, the actions or 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 for implementation of the actions or changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SENS-97-027
REVISION 0

UNIT	:	3 & 4
APPROVAL DATE	:	02/28/97

SAFETY EVALUATION RELATED TO TEMPORARY ALTERATION OF THE "C" GAS DECAY TANK SAMPLE PATH

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with a temporary alteration of the "C" gas decay tank sample path. A temporary alteration of the sample path was proposed by TSA No. 00-97-061-009 because a failure of valve PCV-1038B prevented the normal sample path from being used. The Turkey Point UFSAR requires that the contents of a gas decay tank be sampled prior to release through the monitored plant vent stack. The proposed sample path was from the inlet of PT-1038 to the upstream side of PCV-1073B, which discharges to the waste gas analyzer sample header near the normal sample discharge point. This sample path provided a direct indication of the stored activity in the "C" gas decay tank and was consistent with the normal sampling procedure. To prevent the accidental release of the gas decay tank contents in the proposed configuration, an additional valve was installed in the piping between the temporary tie-in point at the inlet of PCV-1073B, and waste gas release valve RCV-014. The new valve was administratively locked closed in compliance with UFSAR requirements. The tubing run was located within the shielded gas decay tank valve gallery so plant personnel working in the auxiliary building would not be subjected to abnormal dose levels during the temporary system alteration.

Safety Evaluation:

The gas decay tanks and associated discharge piping do not perform any safety related functions. Inadvertent release of the contents of a single tank is bounded by the UFSAR analysis of a gas decay tank rupture. In addition, the UFSAR commitment that two valves remain closed in series to provide a positive means of preventing an inadvertent release of a gas decay tank contents during sampling remains intact during the temporary system alteration. Consequently, 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 or changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEFJ-97-030
REVISION 0

UNIT	:	3 & 4
APPROVAL DATE	:	09/02/97

10 CFR 50.59 SAFETY EVALUATION - EVALUATION FOR PERFORMING THE ROD DROP TEST FROM ALL RODS OUT CONDITION

Summary:

This safety evaluation assess the acceptability of performing the hot rod drop test from an all rods out (ARO) condition. This would reduce the amount of time spent performing low power physics tests after a refueling outage. The current test procedure measured the rod drop times of two banks at a time. Since there are a total of six banks in the Turkey Point core, the test is repeated three times to gather data for all of the control rod banks. The proposed method of testing allows all six control rod banks to be tested at the same time. A review of the UFSAR and plant technical specifications indicated that there were no licensing commitments associated with a particular test sequence or particular test condition. To accommodate the proposed test procedure changes, contingency measures were established to ensure that adequate shutdown margin would be maintained during the ARO condition. These included:

a) a requirement to borate the core to a boron concentration corresponding to a Keff of < 0.99 with uncertainty allowance; and

b) a requirement to perform a 1/M plot at the time the control rods are being withdrawn to the ARO position during the test.

The safety evaluation examined the impact of proposed procedure changes on the plant safety analyses and provided a disposition for each UFSAR Chapter 14 event.

Safety Evaluation:

The safety evaluation demonstrated that the proposed changes in rod drop time testing did not impact the plant technical specifications or any UFSAR safety analyses. Restrictions were imposed to ensure that the minimum required technical specification shutdown margin is maintained during rod withdrawal to the ARO position. Based on the precautions identified, the evaluation concluded that the rod drop test could be performed prior to criticality with Tavg ≥ 541 °F (considered as Mode 3), 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 activities identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEMS-97-033
REVISION 0

UNIT : 3
APPROVAL DATE : 03/28/97

SAFETY EVALUATION FOR UNIT 3 REACTOR CORE FOREIGN OBJECT

Summary:

This safety evaluation was prepared by Westinghouse Electric Corporation to assess the impact on plant safety and operation associated with a loose nylon cable tie in the reactor coolant system. The cable tie came apart while restraining temporary underwater camera and light cables during the refueling outage. The cable tie originally fell to the top of a fuel assembly, but was subsequently lost during retrieval efforts. It was assumed that the missing cable tie was made of white nylon 6/6 based on the material of adjacent cable ties on the camera and light rig. The safety evaluation provided an assessment of the potential impact of the unrecovered cable tie on the operability and integrity of the reactor coolant system (RCS) and interfacing safety-related auxiliary systems during future operating cycles. The assessment considered the potential impact on materials compatibility, fuel integrity, core thermal-hydraulics, core physics characteristics, RCS components, auxiliary components, and instrumentation and control systems.

Safety Evaluation:

The nylon cable tie is expected to soften during heatup and completely melt prior to reaching the normal RCS operating temperature. It is further expected that the melted nylon material will be dispersed into the RCS and deposited on cooler surfaces under low flow conditions. The safety evaluation demonstrated that this phenomenon would not affect the operability or integrity of the reactor fuel, RCS, or the interfacing auxiliary systems. In addition, the decomposition of the nylon material due to extended thermal and radiation exposure would not alter the results of any previously performed radiological dose calculations. Based on this assessment, the evaluation determined that the presence of a single nylon cable tie in the RCS did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for continued operation of the plant.

UNIT : 3 & 4
APPROVAL DATE : 04/07/97

SAFETY EVALUATION FOR FREEZE SEALS ON SEAL TABLE THIMBLE GUIDE TUBE

Summary:

This safety evaluation was prepared to assess the performance and use of freeze seals when conducting repairs on the bottom mounted in-core thimble guide tubes associated with the plant seal table. It was generated in response to an identified through-wall leak in the thimble guide tube associated with core location H-1. To avoid plant operation in a reduced inventory condition, it was proposed that two freeze seals be placed on the thimble guide tube to be repaired to allow reactor coolant system (RCS) inventory to remain in the range of the pressurizer. The activity of placing a freeze seal on a guide tube has previously been evaluated for Turkey Point and found acceptable by Westinghouse Electric Corporation. The intent of this safety evaluation was to review that evaluation and reconfirm its validity for this application. Although the review concluded that the activities proposed for guide tube H-1 were essentially identical to those previously evaluated, implementation of the proposed freeze seals was re-evaluated under the criteria of 10 CFR 50.59.

Safety Evaluation:

This evaluation addressed the consequences of a freeze plug failure (both seals) and demonstrated that sufficient makeup capability could be provided to ensure accomplishment of the decay heat removal function. It was also concluded that adequate precautions have been included in the repair procedure to preclude vertical movement of the thimble tube while the freeze plug is intact. Based on the precautions identified, the evaluation concluded that the maintenance could be performed, and that this 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 activities identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SENS-97-040
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 04/29/97

SAFETY EVALUATION RELATED TO STEAM GENERATOR BLOWDOWN CONTAINMENT ISOLATION FEATURES

Summary:

This safety evaluation was prepared to clarify the function of the steam generator blowdown isolation valves and blowdown sample valves as described in the UFSAR relative to containment isolation. It was prepared in response to an operator discovery that the Unit 3 steam generator blowdown system interlock bypass switches were left in the drain/fill position when the unit entered Mode 4 from a recent refueling outage (LER 250/97-003). The drain/fill position blocks Phase A, main steam isolation, and auxiliary feedwater actuation signals to the main blowdown isolation valves CV-3-6275 A, B, and C, and blowdown sampling valves MOV-3-1425, 1426, and 1427, such that these valves are open and will not close automatically. The evaluation demonstrated that the valves were not containment isolation valves and not subject to the containment isolation technical specification requirements. It was established that the primary function of these valves was to close to support operation of the auxiliary feedwater system, which is required in Modes 1 through 3. The design basis clarification was supported with statements taken from original Westinghouse Electric Corporation design information, and a detailed analysis of the overall secondary system isolation philosophy implemented at Turkey Point. The proposed UFSAR changes were provided as Attachment 1 to the evaluation

Safety Evaluation:

The safety evaluation addressed the impact of the mispositioned keylock switches on the plant safety analyses for Mode 4 events. It was demonstrated that sufficient contingency actions were included in the emergency operating procedures to ensure that manual closure of the blowdown valves would occur if their associated keylock switches were in the "override" position. The evaluation also demonstrated that leaving the subject valves open during plant operation would not invalidate the UFSAR single active failure criterion for containment isolation. It was concluded that the change in interpretation would not reduce the level of protection provided against the release of radioactivity to the outside environment because the subject valves continue to receive an automatic phase A closure signal. Consequently, the 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 proposed document changes.

SAFETY EVALUATION PTN-ENG-SEIS-97-041
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 06/27/97

SAFETY EVALUATION FOR CORE EXIT THERMOCOUPLE ABANDONMENT OR REPLACEMENT

Summary:

This safety evaluation was developed to establish a generic set of guidelines that could be used by Maintenance personnel to abandon or restore core exit thermocouples (CETs), via the plant abandoned equipment program. This would allow inoperable CETs to be abandoned in place (or restored) without performing repetitive engineering evaluations. The CETs are part of the inadequate core cooling system (ICCS) instrumentation which is designed to yield information on fuel assembly outlet temperatures at selected core locations. This information is used to confirm that reactor core design parameters are within analyzed limits. The number of operable CETs per core quadrant is governed by plant technical specifications. In keeping with the applicable requirements, this safety evaluation permitted the abandonment of a given CET if, after the abandonment, at least 2 operable CETs per channel per core quadrant (4 total per quadrant) remained in service. Any abandonment activity that would result in fewer than 2 operable CETs on a given channel and core quadrant was not permitted by the safety evaluation, even though a minimum of 1 operable CET on a given channel and core quadrant was permitted by plant technical specifications.

Safety Evaluation:

The CETs provide a means for acquiring information about the core, but do not perform any type of control function. Since the number of remaining operable CETs continued to satisfy technical specification requirements, there was no safety significance to the proposed abandonment guidelines. The actions or 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 conditions identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SECS-97-051
REVISION 0

UNIT : 4
APPROVAL DATE : 04/25/97

10 CFR·50.59 SAFETY EVALUATION FOR THE PLACEMENT OF LEAD SHIELDING ON VALVE LCV-4-460, AND SHIELD WALL BETWEEN VALVE LCV-4-460 AND REGEN HEAT EXCHANGER

Summary:

This safety evaluation addressed the temporary installation of lead shielding inside containment to permit in-situ repair of the regenerative heat exchanger level control valve, LCV-4-460. The purpose of the evaluation was to demonstrate that the temporary shielding would not adversely affect plant safety or operation if erected in Mode 3 or below. The evaluation examined the potential for adverse seismic interactions between the lead shielding and adjacent safety related equipment, the increased pipe stresses caused by the placement of lead blankets on valve LCV-4-460, the potential impact on containment isolation for penetration No. 19, and the potential impact on containment sump operation during postulated reactor coolant system pipe rupture events.

Safety Evaluation:

The temporary shielding covered by the scope of this safety evaluation was designed to withstand all applicable loads, including seismic loads. The lead blankets and temporary support structure did not modify or actively interact with any plant equipment important to safety. An engineering review demonstrated that containment isolation and containment sump operation would not be affected by the shielding material or the scaffolding support structure. The temporary plant 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 identified within this evaluation.

UNIT : 3
APPROVAL DATE : 05/09/97

SAFETY EVALUATION RELATED TO TEMPORARY REMOVAL OF THE 3A CCW HEAT EXCHANGER CHANNEL HEADS AND TEMPORARY SUPPORT INSTALLATION

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with the temporary removal of the 3A component cooling water (CCW) heat exchanger channel heads for maintenance access to the tube sheets. Access to the inlet and outlet tube sheets was necessary for maintenance personnel to retube the heat exchanger. The CCW heat exchanger inlet and outlet channel heads provide an anchor point for the seismic analysis of the ICW system. A stress analysis of the system without these anchor points determined that the ICW/CCW system would remain operable provided that a temporary support was added to the 3B CCW heat exchanger support pedestal. Removing the channel heads from the heat exchanger shell required that a heavy load lift be performed in a restricted area. To minimize the impact on neighboring equipment, provisions were included in the safety evaluation to lower the channel heads straight down to the 18' elevation where there was no safety related equipment in the vicinity. A review of the Turkey Point UFSAR and plant technical specifications indicated that the CCW system is capable of accomplishing its safety related function with two of the three heat exchangers in service. Thus, plant operation with one CCW heat exchanger out of service for maintenance has been previously reviewed and evaluated.

Safety Evaluation:

The ability to remove one of the three CCW heat exchangers from service for repair or replacement is part of the system design basis. The requirement to install supplemental restraints on the 3A support pedestal preserved the seismic qualification of the system during the maintenance evolution, and ensured continued operability of the 3B and 3C heat exchangers. Consequently, the actions and plant 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 and plant changes identified in this safety evaluation.

SAFETY EVALUATION PTN-ENG-SENS-97-058
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 07/24/97

SAFETY EVALUATION FOR REPEALING THE FIRE BARRIER REQUIREMENTS ASSIGNED TO PENETRATION No. 041E-E001 IN THE EAST WALL OF THE BORIC ACID STORAGE TANK ROOM (FIRE AREA M/41)

Summary:

This safety evaluation addressed the 10 CFR 50.59 criteria for repealing the fire barrier requirements assigned to penetration No. 041E-E001 in the east wall of the Units 3&4 boric acid storage tank room. The east wall is an external wall which separates the boric acid batching equipment from the outside radiation control area yard and the Unit 3 component cooling water (CCW) equipment area. The 4-inch diameter wall penetration is routinely used during refueling outages to refill the boric acid storage tanks from an external supply. Plant drawings and field walkdowns were used to demonstrate that adequate spatial separation existed between the boric acid storage tanks and the refueling water storage tanks (the redundant water source for safe shutdown) to ensure that one borated water source would be available for cooldown during a fire. Administrative controls also existed to limit the presence on intervening combustibles between the two water sources. Several contingency actions were imposed by the safety evaluation to preclude spurious actuation of the 3C/4C ICW and CCW pumps due to a fire in the boric acid storage tank room. Cables associated with these pumps are routed in close proximity to the exterior wall penetration. The repeal of the fire barrier requirement permits the wall penetration to be used without having to install a temporary fire seal, or having to post a fire watch. It also eliminates the need to perform periodic fire barrier related inspections of the penetration sleeve during plant operation. The proposed UFSAR changes were provided in Attachment 1 of the safety evaluation.

Safety Evaluation:

The proposed fire barrier changes did not have any safety significance because safe shutdown during fire scenarios is not a safety related function. Consequently, the actions and 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 and document changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SEIS-97-059
REVISION 0

UNIT : 4
APPROVAL DATE : 07/07/97

SAFETY EVALUATION FOR TEMPORARY INSTALLATION OF A RECORDER ON RCS FLOW LOOP F-4-426 (TSA 04-97-049-3)

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with the temporary installation of a recorder on reactor coolant system (RCS) flow loop F-4-426 for channel troubleshooting purposes. The recorder was installed to monitor the flow comparator input and output signals and the loop power supply, and identify any abnormalities. The recorder was placed on a rubber mat on the floor of the control room, adjacent to rack 4QR15. The evaluation addressed the potential for both electrical and seismic interactions with other safety systems. The recorder was evaluated as a potential missile during a seismic event and determined not represent a hazard for other safety systems. The following electrical failures were also considered: a) loss of power to the recorder, b) shorting of any pair of input leads to each other, c) shorting of any single input lead to any other, d) grounding of any input lead, e) opening of any input lead, and f) incorrect hookup to any in proximity terminal. The failure modes and effects analysis concluded that the temporary installation would not introduce any new failure modes for the flow channel being monitored. Installation of the recorder was limited to a specific period in time after which it was removed under the requirements of the evaluation.

Safety Evaluation:

The evaluation concluded that the installation of the temporary monitoring recorder would have no adverse impact on plant safety or operation, and would not have compromised the safety or licensing requirements for Unit 4. Its installation was also limited to a specific period in time and was to be removed after data was obtained. Consequently, installation of this temporary monitoring recorder, as discussed 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 installation and use of the temporary monitoring recorder.

SAFETY EVALUATION PTN-ENG-SECS-97-061
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 09/04/97

SAFETY EVALUATION FOR CONTAINMENT POLAR CRANE MAINTENANCE INSPECTION PROCEDURE

Summary:

This safety evaluation assessed the acceptability of spreading the required containment polar crane maintenance inspections over the course an outage to reduce critical path time. The current practice of performing all of the required crane inspections at the beginning of each outage delays many of the critical path activities. As a basis for establishing the new inspection schedule, licensing commitments and industry standards were reviewed to determine the minimum set of inspections that were applicable to the polar cranes. The evaluation examined those inspections that had to be performed on a periodic basis and those that had to be performed on a frequent basis (i.e., monthly or daily). The periodic inspection requirements were separated into pre-service activities, preventive maintenance activities, and post-service activities. Pre-service inspections were considered to be valid for one year. Preventive maintenance and post-service inspections were considered to be valid for two years. The identified activities (periodic and frequent) were incorporated into a temporary inspection procedure and scheduled to be performed during the Unit 4 Cycle 17 refueling outage.

Safety Evaluation:

The containment polar cranes do not perform a safety related function so there was no safety significance associated with the proposed activity. Since all licensing commitments were maintained by the proposed inspection plan, the actions or 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 phased polar crane inspection plan.

SAFETY EVALUATION PTN-ENG-SEES-97-062
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 08/06/97

SAFETY EVALUATION FOR SURVEILLANCE REQUIREMENTS FOR APPENDIX R CIRCUITS AT 4.16 KV SWITCHGEAR AND 480 V LOADCENTER BUSSES

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with extending the surveillance requirement for Appendix R circuits at the 4160 V switchgear and 480 V load centers from every refueling outage to every other refueling outage. The Appendix R circuits at the switchgear and load centers consist of transfer switches, breaker control test switches, isolation switches, and redundant fuses. The transfer switches, test switches, and isolation switches used at Turkey Point are either General Electric Type SB-1 switches or Electro-Switch Type 24 switches. A review of the failure history of these switches was performed over the period between July 1990 and July 1997. The review demonstrated that the switches are not subject to any time dependent failure modes. In addition, the subject switches are maintained in a controlled environment which minimizes the probability of contact corrosion and dust accumulation.

Safety Evaluation:

The circuits, switches, and fuses associated with Appendix R do not have any safety significance because safe shutdown during fire scenarios is not a safety related function. Changing the surveillance test frequency of these devices did not adversely impact operation of the plant safety related buses. Consequently, the 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 for implementation of the new surveillance test frequency.

SAFETY EVALUATION PTN-ENG-SEMS-97-065
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 07/29/97

SAFETY EVALUATION TO DOCUMENT THE CHANGE FOR AUXILIARY FEEDWATER SYSTEM STEAM TRAPS ST-33, ST-34, AND ST-35 DRAINING INTO THE ADJACENT TROUGH IN LIEU OF THE EXISTING CONDENSER DRAIN-PATH

Summary:

This safety evaluation was prepared to assess the acceptability of rerouting the ST-33, ST-34, and ST-35 steam trap discharge piping from the existing condenser drain path to an adjacent drain trough. The proposed change was implemented to eliminate potential air in-leakage to the condenser through the steam trap seats. The safety evaluation demonstrated that adequate drain flow would be maintained by the modified piping arrangement. The required UFSAR changes were included as an attachment to the safety evaluation.

Safety Evaluation:

The design change did not adversely impact operation of the auxiliary feedwater system (AFW), access to AFW components, or habitability in the AFW equipment area. Consequently, the plant changes and 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-97-066
REVISION 1

UNIT : 3 & 4
APPROVAL DATE : 08/21/97

SAFETY EVALUATION FOR AUGMENTED RCP OIL COLLECTION SYSTEM REQUIREMENTS

Summary:

This safety evaluation addressed the 10 CFR 50.59 criteria for the proposed reactor coolant pump (RCP) oil collection system changes inside containment. The proposed changes extended the coverage of the existing oil collection system to include a) the upper and lower oil reservoir level switch assemblies and associated piping flanges and drains, b) the oil cooler piping drain, c) the upper oil reservoir drain, and d) the two flexible oil fill line hose connections currently being installed by PC/M 97-016. These changes were needed to comply with the 10 CFR 50 Appendix R requirement that all potential pressurized and unpressurized leakage sites be provided with collection facilities. Only those potential leak sites located below the normal reservoir oil fill level were considered within the scope of the modification. The safety evaluation required that new drip pans and drain lines be installed to collect all dripping and stream-flow type leaks postulated to occur from the identified sources. The new oil collection facilities were required to be seismically installed to prevent adverse interactions with adjacent safety related equipment. In addition, drip pans were required to be mounted to the RCP motor casings at locations that did not provide a fluid or lube oil pressure boundary function; did not house sensitive motor components; and did not compromise electrical insulation requirements of the motor housing configuration. The proposed UFSAR changes were provided as Attachment 1.

Revision 1 documented the Westinghouse position that the main oil reservoir flange on the motor housing was not considered to be a potential source of lube oil leakage.

Safety Evaluation:

The RCP oil collection function is a plant fire protection feature and has no safety significance. The additional drip pans and drain lines required to augment the existing oil collection facilities were determined to have a negligible impact on the containment heat sink analysis. Consequently, the actions and documentation 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 and document changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SENS-97-076
REVISION 0

UNIT : 4
APPROVAL DATE : 09/08/97

SAFETY EVALUATION FOR TEMPORARY ALTERATION OF CONTAINMENT PURGE VALVE POV-4-2600 CONTROL AND INDICATION CIRCUIT WIRING

Summary:

This safety evaluation assessed the impact on plant safety and operation associated with a temporary wiring change in the containment purge valve POV-4-2600 control and indication circuit, to restore valve operability. The temporary wiring change was necessary to circumvent a damaged cable in the valve control circuitry and permit operation of the containment purge system in preparation for an impending unit shutdown and refueling outage. The modification consisted of abandoning the faulty cable in place and rewiring valve POV-4-2600 to the control circuit of valve POV-4-2602. Valve POV-4-2600 is the outboard containment isolation valve for the purge supply line. Valve POV-4-2602 is the outboard containment isolation valve for the purge exhaust line. The circuits were rewired such that both valves could be operated with the POV-4-2602 control switch. The open and close position indications for POV-4-2602 were also rewired in series with POV-4-2600 to maintain position indication for POV-4-2600 in compliance with Regulatory Guide 1.97 commitments. In keeping with these changes, the indication for POV-4-2602 represented the position of both valves, POV-4-2600 and POV-4-2602. The evaluation addressed the applicable containment isolation design bases, the potential for new failure modes, the continuous load rating of the POV-4-2602 control switch, and the impact of the indicating lamp changes on IST testing.

A tabulation of the various indicating light combinations relative to valve POV-4-2600 and POV-4-2602 positions was provided as an attachment to the evaluation for operator training purposes.

Safety Evaluation:

This evaluation examined the potential for new failure modes. It was concluded that the circuit changes did not alter the method of isolating the containment purge lines during an accident, or retard the closing speed of the containment purge valves. The UFSAR commitment that containment isolation be established assuming a single active failure remained intact with the modified design. Consequently, the temporary plant 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 circuit changes.

SAFETY EVALUATION PTN-ENG-SECS-97-077
REVISIONS 0 AND 1

UNIT	:		4
APPROVAL DATE	:	Rev.0	10/02/97
		Rev.1	10/10/97

EVALUATION FOR 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 1 addressed the storage of some additional items inside containment.

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-SEMS-97-078
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 10/14/97

SAFETY EVALUATION FOR CONTINUOUS FIRE WATCH

Summary:

This safety evaluation was prepared to assess the acceptability of performing a continuous fire watch via a 15 minute roving patrol, and to incorporate this definition of a continuous fire watch into the UFSAR so that it accurately reflects the Turkey Point fire protection program. Currently, there is no regulatory definition of a fire watch and none of the NRC fire protection guidance documents provide an explanation of acceptable implementation of a continuous fire watch. The definition established in the safety evaluation requires that a trained individual be in the specified area at all times, that the specified area contain no impediment to restrict the movements of the continuous fire watch, and that each location within a specified area be patrolled at least once every 15 minutes with a margin of 5 minutes. In keeping with these restrictions, the safety evaluation justifies that the implementation of a continuous fire watch via 15 minute roving patrols is an effective utilization of manpower, provides a level of protection that is commensurate with the impaired protective feature, and is capable of detecting a fire before it develops beyond the incipient stage sufficient to cause damage which might affect the ability to achieve and maintain safe shutdown conditions. The proposed UFSAR changes were provided as an attachment to the safety evaluation.

Safety Evaluation:

The implementation of a continuous fire watch does not have any safety significance because safe shutdown during fire scenarios is not a safety related function. Consequently, the actions and 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 and changes identified within this evaluation.

SAFETY EVALUATION PTN-ENG-SENS-97-084
REVISION 0

UNIT : 3 & 4
APPROVAL DATE : 10/02/97

SAFETY EVALUATION FOR CONTAINMENT SUMP SCREEN DESIGN REQUIREMENTS

Summary:

This safety evaluation was prepared to augment the documentation of design basis and performance requirements for the containment sump screens, to provide a more mechanistic basis for assessing sump screen operability, to quantify the ECCS heat sink and hydrogen generating material bulk inventory contribution of the sump screens and to provide for clarifying the UFSAR accordingly. This effort was prompted by the discovery of a hole and various gaps in the containment sump screens during the Unit 4 Cycle 17 refueling outage inspection activities, and during a Unit 3 containment entry at power. The conditions were determined to be outside the Turkey Point design basis as stated in the UFSAR and immediately reported to the NRC (LER 250/97-008).

The evaluation developed a mechanistic basis for assessing Turkey Point sump screen operability by determining the types and quantities of debris likely to be generated and transported to the screens as a result of a design basis accident. The methodology determined the areas of exposure to a credible pipe break and characterized the potential debris based on reviewing design and procurement documents and inspecting as-built installations in the field. Appropriate revisions to the UFSAR were provided in Attachment 1 to the safety evaluation. Attachment 2 provided an analysis of sump screen debris and emergency core cooling system performance relative to the design of the debris resistant fuel assemblies installed in the Turkey point cores. Attachment 3 provided an independent assessment of the likelihood of debris generated by a pipe break in the Turkey Point containment bypassing the as-found containment sump screens.

Safety Evaluation:

The evaluation and supporting analyses demonstrated that the extent of screen clogging described in the UFSAR is non-mechanistic, not credible and extremely conservative based on the type and quantity of debris likely to be generated. It was concluded that the proposed UFSAR clarification, and associated augmentation of screen inspection requirements, did not alter the design or operation of any safety related equipment. 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.

SAFETY EVALUATION SECL-97-200
REVISION 0

UNIT : 4
APPROVAL DATE : 09/25/97

REACTOR COOLANT SYSTEM LOOSE PARTS EVALUATION

Summary:

This safety evaluation was prepared by Westinghouse Electric Corporation to assess the impact on plant safety and operation associated with two loose cap screws and locking cups in the reactor coolant system. The cap screws and locking cups were found missing during an inspection of the 4B reactor coolant pump (RCP) diffuser adapter plate during the last Unit 4 refueling outage. For the purposes of analysis, it was assumed that each RCP lost two cap screws and locking cups in the reactor coolant system. The safety evaluation provided an assessment of the potential impact of the unrecovered parts on the operability and integrity of the fuel, reactor vessel, reactor internals, pressurizer, reactor coolant pump, steam generator, auxiliary equipment, or thermowells during future operating cycles.

Safety Evaluation:

An evaluation of equipment important to safety was performed with the loose parts present in the RCS and it was determined that continued operation of the plant was acceptable. All credible scenarios of migration, lodging, or impacting of the loose objects were considered. It was also demonstrated that continued operation of an RCP with two diffuser adaptor cap screws missing would not adversely impact the design or performance of the RCP. Based on this assessment, the evaluation determined that the presence of up to six cap screws and six locking cups in the RCS did not constitute an unreviewed safety question or require changes to the plant technical specifications. Therefore, prior NRC approval was not required for continued operation of the plant.

SECTION 3

RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 96-024

UNIT : 3
TURN OVER DATE : 11/13/96

TURKEY POINT UNIT 3 CYCLE 15 RELOAD DESIGN (THERMAL POWER UPRATE)

Summary:

This Engineering Package provided the safety evaluation, instructions, and data necessary to operate the Unit 3 Cycle 15 core up to 2300 MWt, as part of the thermal power uprate project. The only design change for the Cycle 15 core was the change in thermal design power from 2200 MWt to 2300 MWt. The normal Tavg remained at 574.2 °F. The data and instructions provided included:

a) INCORE-3D detector constants needed to monitor the incore power distribution for compliance with the limits specified in the Core Operating Limits Report (COLR);

b) operational data supporting expected plant evolutions throughout the cycle, including spent fuel pool heatup rates and reactor core decay heat generation; and

c) COLR parameters for the axial flux difference operating band, K(z) curve, rod insertion limits, and peaking factor Limits.

Key safety parameters associated with Unit 3 Cycle 15 thermal uprate were provided as an attachment to the Engineering Package. Associated UFSAR changes were provided as part of PC/M 96-022, Revision 0.

Safety Evaluation:

The Unit 3 Cycle 15 thermal uprate core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric Corporation, using NRC approved methodology. The uprated core met all applicable design criteria and pertinent licensing bases. The reload analyses demonstrated that operation at the higher power level would not exceed any core design criteria, nor cause the core to operate in excess of pertinent design basis operating limits for the key safety parameters. Demonstrated adherence to applicable standards and acceptance criteria precludes new risks to components and systems. Since provisions for power escalation to uprate conditions and associated documentation changes have no adverse affect on plant safety, security, or operation, the changes addressed 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 of the Cycle 15 (thermal uprate) core reload.

PLANT CHANGE/MODIFICATION 96-025

UNIT : 4
TURN OVER DATE : 11/13/96

TURKEY POINT UNIT 4 CYCLE 16 RELOAD DESIGN (THERMAL POWER UPRATE)

Summary:

This Engineering Package provided the safety evaluation, instructions, and data necessary to operate the Unit 4 Cycle 16 core up to 2300 MWt, as part of the thermal power uprate project. The only design change for the Cycle 16 core was the change in thermal design power from 2200 MWt to 2300 MWt. The normal Tavg remained at 574.2 °F. The data and instructions provided included:

a) INCORE-3D detector constants needed to monitor the incore power distribution for compliance with the limits specified in the Core Operating Limits Report (COLR);

b) operational data supporting expected plant evolutions throughout the cycle, including Spent Fuel Pool heatup rates and reactor core decay heat generation; and

c) COLR parameters for the axial flux difference operating band, K(z) curve, rod insertion limits, and peaking factor limits.

Key safety parameters associated with Unit 4 Cycle 16 thermal uprate were provided as an attachment the Engineering Package. Associated UFSAR changes were provided as part of PC/M 96-022, Revision 0.

Safety Evaluation:

The Unit 4 Cycle 16 thermal uprate core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric Corporation, using NRC approved methodology. The uprated core met all applicable design criteria and pertinent licensing bases. The reload analyses demonstrated that operation at the higher power level would not exceed any core design criteria, nor cause the core to operate in excess of pertinent design basis operating limits for the key safety parameters. Demonstrated adherence to applicable standards and acceptance criteria precludes new risks to components and systems. Since provisions for power escalation to uprate conditions and associated documentation changes have no adverse affect on plant safety, security, or operation, the changes addressed 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 of the Cycle 16 (thermal uprate) core reload.

PLANT CHANGE/MODIFICATION 96-071

UNIT : 3
TURN OVER DATE : 05/07/97

TURKEY POINT UNIT 3 CYCLE 16 RELOAD DESIGN

Summary:

This Engineering Package provided the reload core design for the Turkey Point Unit 3 Cycle 16 reload. The primary design change to the core for Cycle 16 was the replacement of 60 irradiated assemblies with 60 fresh Optimized Fuel Assembly (OFA) Region 18 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 Cycle 16 core was designed to operate at the uprated power level of 2300 MWt. The maximum fuel enrichment was 4.2 w/o which was the first use of fuel with an enrichment above 4.0 w/o at Turkey Point.

Region 18 used the same Debris Resistant Fuel Assembly (DRFA) design as the prior Region 17, except that a Composite Top Nozzle (CAST) assembly was used on the new bundles to reduce component parts, and grooved end plugs were used on the new fuel rods to improve end plug welding. Neither of these manufacturing-related design changes had any impact on fuel performance.

Cross core fuel bundle shuffles were utilized in the Cycle 16 loading pattern; these shuffles were adequate to minimize potential power asymmetries. The fuel was arranged in a low leakage pattern with no significant differences between the Cycle 15 and Cycle 16 loading patterns.

Safety Evaluation:

The Unit 3 Cycle 16 reload core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric Corporation. The Cycle 16 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 16 core reload did not have any adverse effect on plant safety or plant operations. Consequently, the Cycle 16 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 97-014

UNIT : 4
TURN OVER DATE : 10/22/97

TURKEY POINT UNIT 4 CYCLE 17 RELOAD DESIGN

Summary:

This Engineering Package provided the Turkey Point Unit 4 Cycle 17 reload core design. The primary design change for Cycle 17 was the replacement of 56 irradiated assemblies with 56 fresh Optimized Fuel Assembly (OFA) Region 19 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 Cycle 17 core was designed to operate at the uprated power level of 2300 MWt. The maximum fuel enrichment was 4.2 w/o which was the first use of fuel with an enrichment above 4.0 w/o in the Unit 4 core. A Composite Top Nozzle (CAST) assembly was included on the new region 19 bundles to reduce component parts, and a standardized top grid bulge location was established to reduce fuel manufacturing costs. Grooved end plugs were also used on the new fuel rods to improve end plug welding.

Cross core fuel bundle shuffles were utilized in the Cycle 17 loading pattern; these shuffles were adequate to minimize potential power asymmetries. The fuel was arranged in a low leakage pattern with no significant differences between the Cycle 16 and Cycle 17 loading patterns. The plant safety analyses for Cycle 17 incorporated a) the pressure losses in the steam piping between the steam generators and the main steam safety valves, and b) an increased time delay in the turbine pressure signal to the automatic rod control system.

Safety Evaluation:

The Unit 4 Cycle 17 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 accident reanalysis demonstrated that the safety limits continue to be met for Cycle 17. Consequently, the Cycle 17 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 of the Cycle 16 core reload.

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).

Two power operated relief valve (PORV) challenges occurred for the Turkey Point Plant Units 3 and 4 during the period from April 8, 1996 through October 13, 1997.

Unit 3

A PORV actuation occurred when the 3A reactor coolant pump was started for a one minute run during the reactor coolant system fill and vent process at the end of the Cycle 16 refueling outage. The PORVs were operating in the overpresure mitigating system (OMS) mode when the actuation occurred. The OMS operated as designed with PORV (or PORVs) lifting at about 415 psig. A special report for this event was submitted to the NRC Regional Administrator (Region II) under FPL letter L-97-102.

Unit 4

A PORV actuation occurred on April 23, 1997 during a reactor trip from 100% power. Two PORVs lifted during the transient at 2335 psig.

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 4

EXAMINATION DATE: September 18, 1997 thru September 22, 1997

STEAM GENERATOR	TOTAL TUBES INSPECTED	TOTAL TUBES		TUBES PLUGGED AS PREVENTIVE MAINTENANCE	TUBES PLUGGED THIS OUTAGE	TOTAL PLUGGED TUBES IN S/G
		20% - 39%	>40% & VOL			
4E210A	3196	0	0	0	0	16
4E210B	3206	2 (1)	0	0	0	8
4E210C	3205	4 (1)	0	0	0	9

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

STEAM GENERATOR	AVB BARS	SUPPORT LOCATIONS 1 THROUGH 6		FREESPAN 6H thru 6C	TOP OF TUBESHEET TO 1st SUPPORT		TOTAL NUMBER OF INDICATIONS	
		COLD LEG	HOT LEG	U-BEND	COLD LEG	HOT LEG	20% - 39%	>40% & VOL
4E210A	0	0	0	0	0	0	0	0
4E210B	2 (1)	0	0	0	0	0	2 (1)	0
4E210C	4 (1)	0	0	0	0	0	4 (1)	0

Remarks:

(1) Mechanical wear damage at Anti-vibration Bars (AVB) was depth sized using qualified bobbin coil sizing technique.

Date: 10/7/97 Prepared by: [signature]
S/G Eddy Current Coordinator

Date: 10/13/97 Reviewed by: [signature]
CSI Inspections Supervisor

Date: 10/13/97 Reviewed by: Gary L Boyers
CSI S/G Technical Specialist

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 4
09/97

COMPONENT : S/G A

Page : 1
Date : 09/24/

Examination Dates : 09/18/97 thru 09/22/97

Total Number of Tubes Inspected .....: 3198

Total Indications
Between 20% and 39% ....: 0
Greater than or equal to 40% ....: 0
Total Corrosion type Indications "VOL" 0

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 0

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	0	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	0	01C -2.0 to 06C +2.0 :	0
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	0		

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 4
09/97

COMPONENT : S/G B

Page : 1
Date : 09/24/

Examination Dates : 09/18/97 thru 09/22/97

Total Number of Tubes Inspected .....: 3206

Total Indications
Between 20% and 39% ....: 2
Greater than or equal to 40% ....: 0
Total Corrosion type Indications "VOL" 0

Total Tubes Plugged as Preventive Maint.: 0
Total Tubes Plugged ....: 0

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	0	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	0	01C -2.0 to 06C +2.0 :	0
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	2		

INDICATIONS/TRENDING REPORT
PTN-4
OUTAGE : 09/97

COMPONENT : S/G B
DESCRIPTION : 20% to 39% Indications

Page : 1
Date : 10/10/97
Time : 08:29:51

Row	Col	Leg	***	Extent Tst/Note	Reel	Probe	09/97 Location	Volts	Deg	Ch	%	Diff	N/A Location	Volts	Deg	Ch	%
23	50	C		TEH PS	8C044	A-720-M/ULC	AV3 -.2	.5		P 2	26						
30	65	C		TEH PS	8C033	A-720-M/ULC	AV2 -.1	.6		P 2	20						

Number of RECORDS Selected from Current Outage : 2
Number of TUBES Selected from Current Outage : 2

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT ≠ 4
09/97

COMPONENT : S/G C

Page : 1
Date : 09/24

Examination Dates : 09/18/87 thru 09/22/97

Total Number of Tubes Inspected .....: 3205

Total Indications
Between 20% and 39% ....: 4
Greater than or equal to 40% ...: 0
Total Corrosion type Indications "VOL" 0

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....: 0

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	0	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	0	01C -2.0 to 06C +2.0 :	0
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	4		

INDICATIONS/TRENDING REPORT
PTN-4
OUTAGE : 09/97

COMPONENT : S/G C
DESCRIPTION : 20% to 39% Indications

Page : 1
Date : 10/10/97
Time : 08:30:42

				Extent				09/97					N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
22	7	C		TEH SC	CC022	A-720-M/ULC	AV2 1.9	.3		P 2	21						
32	16	C		TEH PS	CC024	A-720-M/ULC	AV2 .0	.3		P 2	22						
35	31	C		TEH PS	CC043	A-720-M/ULC	AV2 .0	.4		P 2	21						
13	43	H		TEC SS	CH001	A-720-M/ULC	AV3 -.9	1.0		P 2	30						

Number of RECORDS Selected from Current Outage : 4
Number of TUBES Selected from Current Outage : 4

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 13, 1997 thru MARCH 19, 1997

STEAM GENERATOR	TOTAL TUBES INSPECTED	TOTAL TUBES		TUBES PLUGGED AS PREVENTIVE MAINTENANCE	TUBES PLUGGED THIS OUTAGE	TOTAL PLUGGED TUBES IN S/G
		20% - 39%	≥ 40% & VOL			
3E210A	3197	5(1)	2	1	3	20
3E210B	3196	4(1)	7	2	9	27
3E210C	3181	13(1)	1	1	2	35

LOCATION OF INDICATIONS

(20% - 100% & VOL)

STEAM GENERATOR	AVB BARS	SUPPORT LOCATIONS 1 THROUGH 6		FREESPAN 6H thru 6C	TOP OF TUBE SHEET TO #1 SUPPORT		TOTAL INDICATIONS	
		COLD LEG	HOT LEG	UBEND	C/L	H/L	20% - 39%	≥ 40% & VOL
3E210A	6(1)	0	0	1	0	1	6(1)	2
3E210B	7(1)	1	1	0	0	8(2)	7(1)	10(2)
3E210C	20(1)	0	0	0	0	1	20(1)	1

Remarks:

(1) Mechanical wear damage at Anti-vibration Bars (AVB) was depth sized using qualified Bobbin Coil Sizing Technique.

(2) Four tubes plugged due to Loose Part Damage.

DATE 5/13/97 PREPARED BY C. Montalbano
S/G EDDY CURRENT COORDINATOR

DATE: 5/13/97 REVIEWED BY: [signature]
INSPECTIONS SUPERVISOR

DATE: 5/13/97 REVIEWED BY: C. Montalbano
S/G TECHNICAL SPECIALIST for E.L.B.

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
03/97

COMPONENT : S/G A

Page .: 1
Date : 05/1

Examination Dates : 03/13/97 thru 03/19/97

Total Number of Tubes Inspected .....: 3197

Total Indications
Between 20% and 39% ....: 6 (1)
Greater than or equal to 40% ...: 0
Total Corrosion type Indications "VOL" 2

Total Tubes Plugged as Preventive Maint : 1
Total Tubes Plugged ....: 3

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	1	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	1	01C -2.0 to 06C +2.0 :	0
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	6 (1)		

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 03/97

COMPONENT : S/G A

DESCRIPTION : ALL 20% TO 39% INDICATIONS

Page : 1
Date : 5/12/97
Time : 10:00:00

				Extent			03/97						N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
33	41	C		TEH PS	AC006	A-720-M/ULC	AV1 .0	.9		P 2	24						
		C		TEH PS	AC006	A-720-M/ULC	AV3 .0	.9		P 2	25						
38	45	C		TEH PS	AC007	A-720-M/ULC	AV2 .0	1.5		P 2	30						
37	47	C		TEH PS	AC012	A-720-M/ULC	AV3 .0	.8		P 2	22						
30	52	H		TEC PC	AH034	A-720-M/ULC	AV3 .0	.4		P 2	20						
28	59	H		TEC PS	AH021	A-720-M/ULC	AV2 .0	.6		P 2	26						

Number of RECORDS Selected from Current Outage : 6

Number of TUBES Selected from Current Outage : 5

INDICATIONS/TRENDING REPORT

PTN-3

OUTAGE : 03/97

COMPONENT : S/G A

DESCRIPTION : ALL 40% to 100%, VOL & PTP INDICATIONS

Page : 1

Date : 5/12/97

Time : 10:00:00

				Extent			03/97						N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
13	5	H	CHR	BAHTSHPL	AH002	MRPC680-3C	TSH 3.8	1.4	119	1	PTP						
31	18	H		06H SC	AH039	MPRC720-3C	06H 1.1	1.0	64	1	VOL						
44	36	C			AH037	MPRC720-3C					PTP						
		C	CHR	TSH GY	AH037	MPRC720-3C	TSH .7	1.2	14	1	VOL						

Number of RECORDS Selected from Current Outage : 4

Number of TUBES Selected from Current Outage : 3

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
03/97

COMPONENT : S/G B

Page : 1
Date : 05/12

Examination Dates : 03/13/97 thru 03/19/97

Total Number of Tubes Inspected .....: 3196

Total Indications	
Between 20% and 39% ....:	7 (1)
Greater than or equal to 40% ....:	1
Total Corrosion type Indications "VOL"	9 (2)
Total Tubes Plugged as Preventive Maint :	2
Total Tubes Plugged ....:	9 (2)

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	8	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	1	01C -2.0 to 06C +2.0 :	1
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	7 (1)		

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 03/97

COMPONENT : S/G B
DESCRIPTION : ALL 20% TO 39% INDICATIONS

Date : 5/12/97
Time : 10:00:00

				Extent				03/97					N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
32	34	C		TEH PS	BC019	A-720-M/ULC	AV1 .0	.7		P 2	23						
		C		TEH PS	BC019	A-720-M/ULC	AV3 -.1	1.3		P 2	31						
		C		TEH PS	BC019	A-720-M/ULC	AV4 -.2	.7		P 2	22						
34	46	C		TEH PC	BC009	A-720-M/ULC	AV2 .0	1.0		P 2	28						
		C		TEH PC	BC009	A-720-M/ULC	AV3 .0	1.5		P 2	35						
34	51	C		TEH PS	BC011	A-720-M/ULC	AV2 .0	1.2		P 2	29						
42	55	C		TEH PC	BC012	A-720-M/ULC	AV3 -.3	1.7		P 2	31						

Number of RECORDS Selected from Current Outage : 7
Number of TUBES Selected from Current Outage : 4

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 03/97

COMPONENT : S/G B
DESCRIPTION : ALL 40% to 100%, VOL & PTP INDICATIONS

Date : 5/12/97
Time : 10:00:00

				Extent				03/97					N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
37	20	H	CMR	TSH PL	BH002	MRPC680-3C	TSH .4	1.0	102	1	VOL						
42	38	H	CMR	01HTSHPL	BH004	MRPC680-3C	TSH 1.5	1.2	87	1	VOL						
39	39	C	CMR	05H PL	BH043	MRPC680-3C	05H .8	2.3	125	1	VOL						
44	40	C	CMR	TSH PL	BH043	MRPC680-3C	TSH .3	1.3	46	1	VOL						
		C	PIT	TSH PL	BH043	MRPC680-3C	TSH 1.8	.4	60	P1	82						
23	41	C	CMR	03C PC	BC031	MRPC680-3C	03C .5	.4	93	1	VOL						
44	41	C	CMR	TSH PL	BH043	MRPC680-3C	TSH .2	1.5	122	1	VOL						
		C	CMR	TSH PL	BH043	MRPC680-3C	TSH .4	1.1	90	1	VOL						
		C	CMR	TSH PL	BH043	MRPC680-3C	TSH .5	.9	83	1	VOL						
45	41	C	LPM	TEH PL	BC009	A-720-M/ULC	TSH .6				PTP						
45	42	C	LPM	TEH PL	BC009	A-720-M/ULC	TSH 1.3				PTP						
41	44	C	CMR	TEH GY	BH043	MRPC680-3C	TSH .6	.3	52	1	VOL						

Number of RECORDS Selected from Current Outage : 12
Number of TUBES Selected from Current Outage : 9

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
03/97

COMPONENT : S/G C

Page : 1
Date : 05/1

Examination Dates : 03/13/97 thru 03/19/97

Total Number of Tubes Inspected .....: 3181

Total Indications
Between 20% and 39% ....: 20 (1)
Greater than or equal to 40% ...: 1
Total Corrosion type Indications "VOL" 0

Total Tubes Plugged as Preventive Maint : 1
Total Tubes Plugged ....: 2

Location Of Indications 20% to 100% & "VOL"

Hot Leg		Cold Leg	
TSH -.5 to 01H -2.1 :	1	TSC -.5 to 01C -2.1 :	0
01H -2.0 to 06H +2.0 :	0	01C -2.0 to 06C +2.0 :	0
06H +2.1 to AV1 -3.1 :	0	06C +2.1 to AV4 -3.1 :	0
AV1 -3.0 to AV4 -3.0 :	20 (1)		

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 03/97

COMPONENT : S/G C
DESCRIPTION : ALL 20% TO 39% INDICATIONS

Date : 5/12/97
Time : 10:00:00

				Extent				03/97					N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
30	17	H		TEC PS	CH006	A-720-M/ULC	AV1 .0	.5		P 2	20						
34	31	H		TEC PS	CH027	A-720-M/ULC	AV2 .0	.5		P 2	21						
		H		TEC PS	CH027	A-720-M/ULC	AV3 .0	.5		P 2	20						
43	33	H		TEC PS	CH026	A-720-M/ULC	AV3 .0	.4		P 2	22						
35	36	H		TEC PS	CH028	A-720-M/ULC	AV2 .0	.5		P 2	23						
34	41	H		TEC PS	CH029	A-720-M/ULC	AV1 .0	.6		P 2	22						
		H		TEC PS	CH029	A-720-M/ULC	AV3 .0	.6		P 2	22						
		H		TEC PS	CH029	A-720-M/ULC	AV4 .0	.7		P 2	24						
33	43	H		TEC PC	CH030	A-720-M/ULC	AV3 .0	.6		P 2	27						
35	43	H		TEC PC	CH030	A-720-M/ULC	AV2 .0	.8		P 2	21						
		H		TEC PC	CH030	A-720-M/ULC	AV3 .0	1.4		P 2	32						
		H		TEC PC	CH030	A-720-M/ULC	AV4 .0	.9		P 2	23						
35	44	H		TEC PC	CH030	A-720-M/ULC	AV2 .0	1.5		P 2	34						
		H		TEC PC	CH030	A-720-M/ULC	AV3 .0	1.6		P 2	35						
30	48	H		TEC	CH009	A-720-M/ULC	AV2 .0	1.1		P 2	32						
		H		TEC PC	CH009	A-720-M/ULC	AV3 .0	1.5		P 2	37						
35	49	C		TEH PS	CC009	A-720-M/ULC	AV4 .0	.6		P 2	25						
26	58	H		TEC	CH009	A-720-M/ULC	AV2 .0	.5		P 2	20						
30	61	H		TEC PS	CH009	A-720-M/ULC	AV2 .0	.5		P 2	20						
38	71	C		TEH PS	CC011	A-720-M/ULC	AV4 -.2	.6		P 2	23						

Number of RECORDS Selected from Current Outage : 20
Number of TUBES Selected from Current Outage : 13

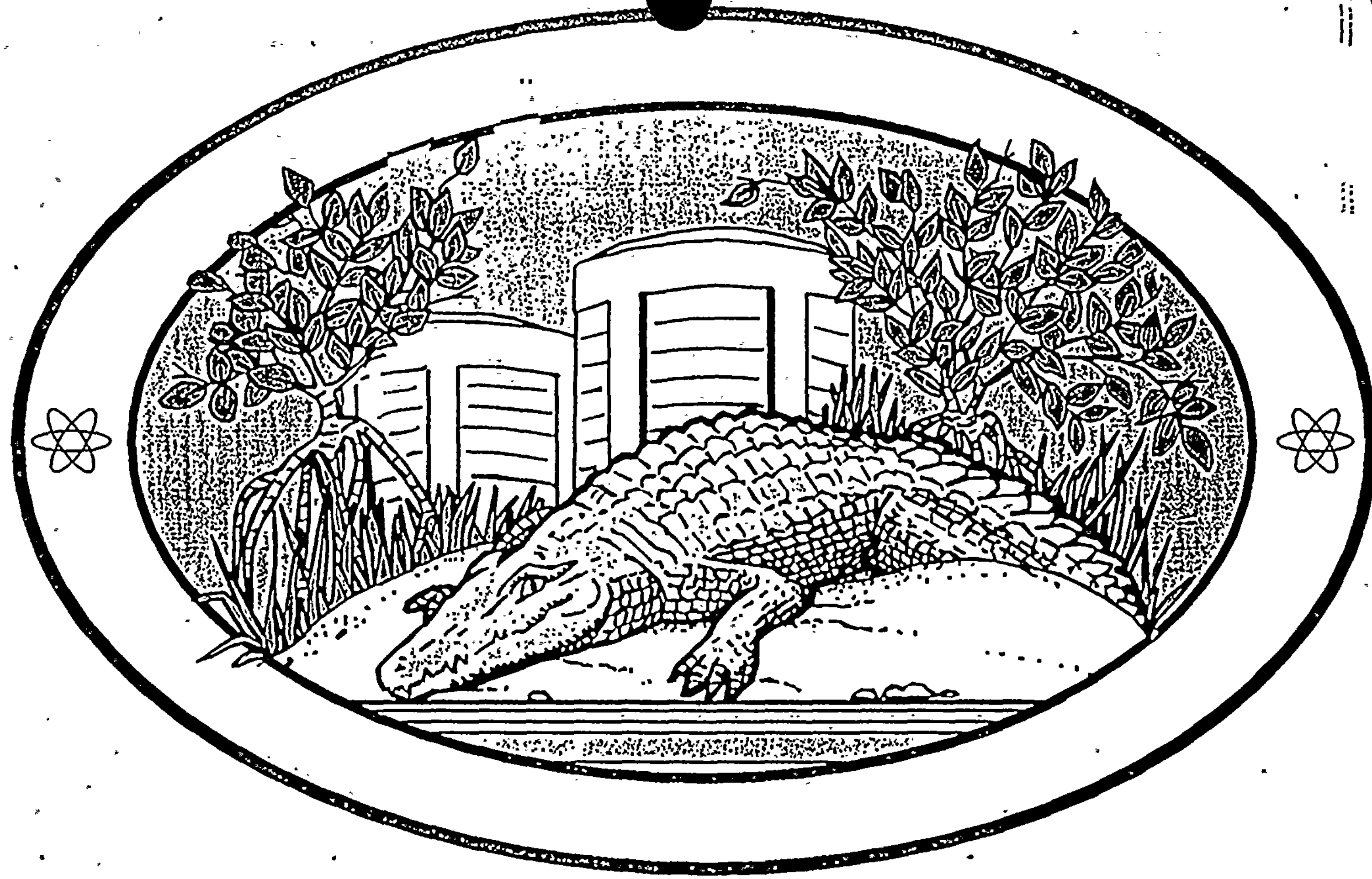
INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 03/97

COMPONENT : S/G C
DESCRIPTION : ALL 40% to 100%, VOL & PTP INDICATIONS

Page : 1
Date : 5/12/97
Time : 10:00:00

				Extent			03/97						N/A				
Row	Col	Leg	***	Tst/Note	Reel	Probe	Location	Volts	Deg	Ch	%	Diff	Location	Volts	Deg	Ch	%
30	48	H		TEC PL	CH009	A-720-M/ULC	AV3 .0				PTP						
33	66	C	PIT	TSH PL	CH047	MRPC680-3C	TSH .6	.4	103	P 1	58						

Number of RECORDS Selected from Current Outage : 2
Number of TUBES Selected from Current Outage : 2



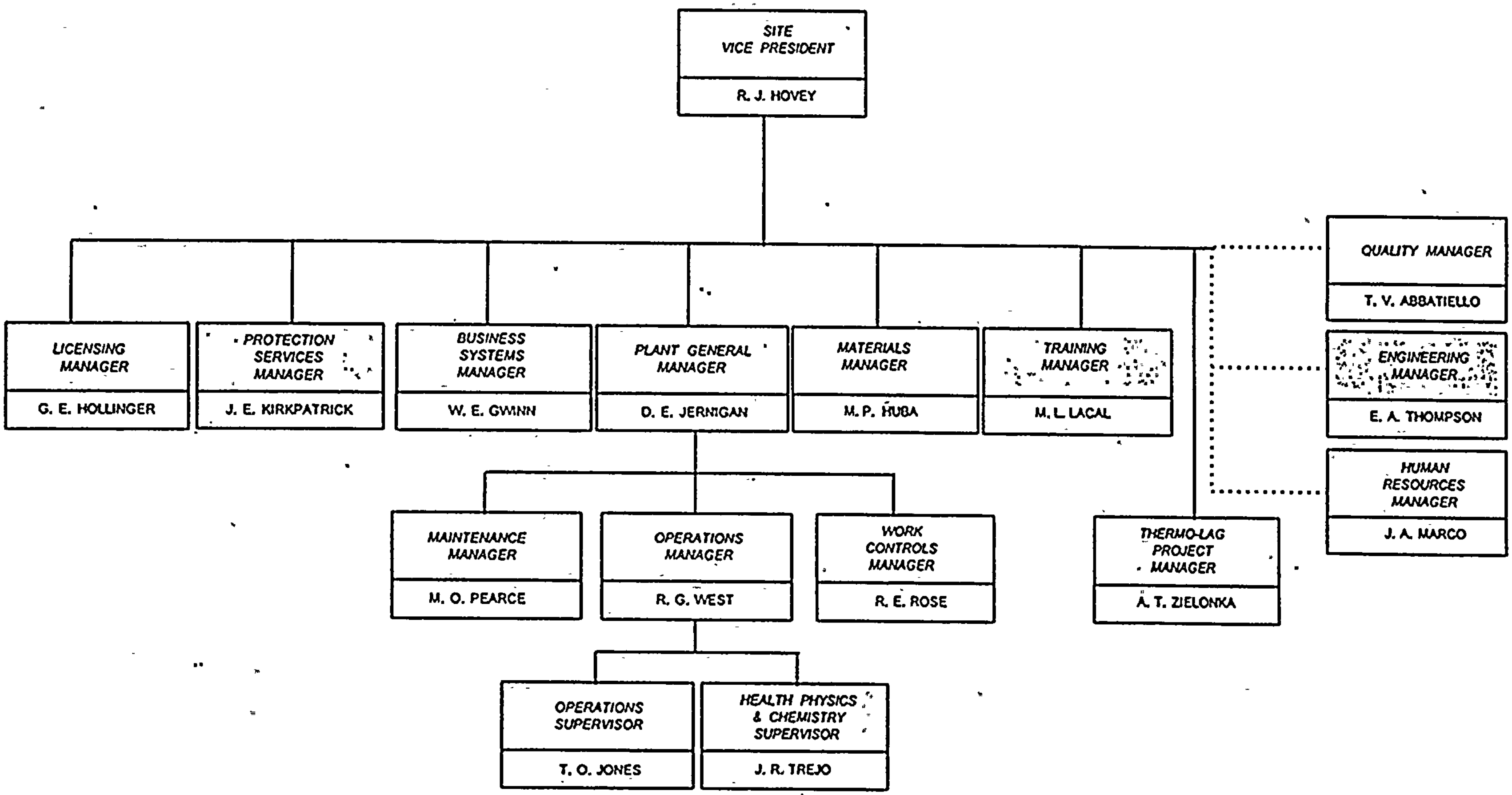
Turkey Point Nuclear Plant
Presentation to the NRC
May 12, 1998
Atlanta, Ga.

Agenda

INTRODUCTION

R. J. Hovey

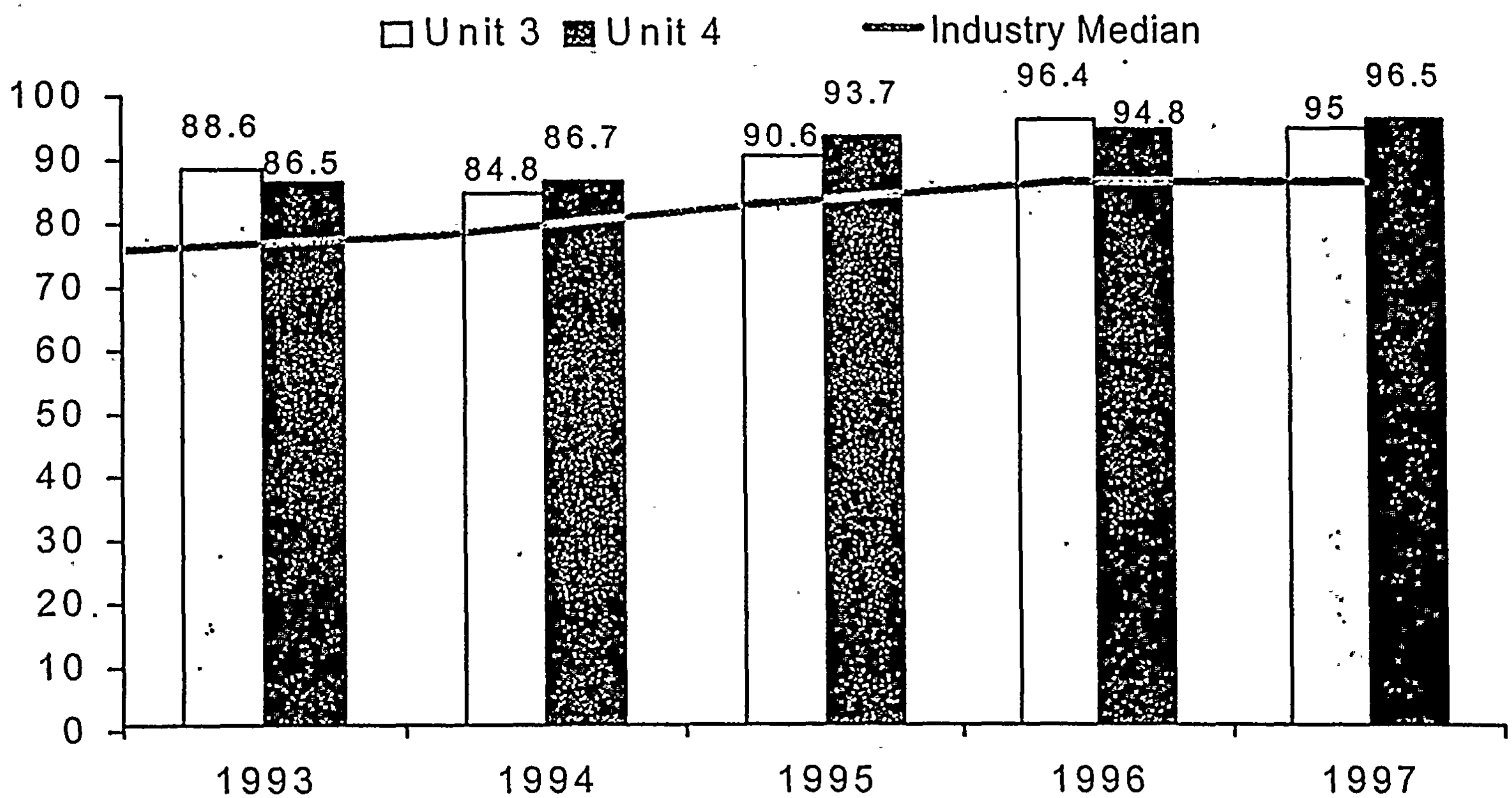
ORGANIZATION



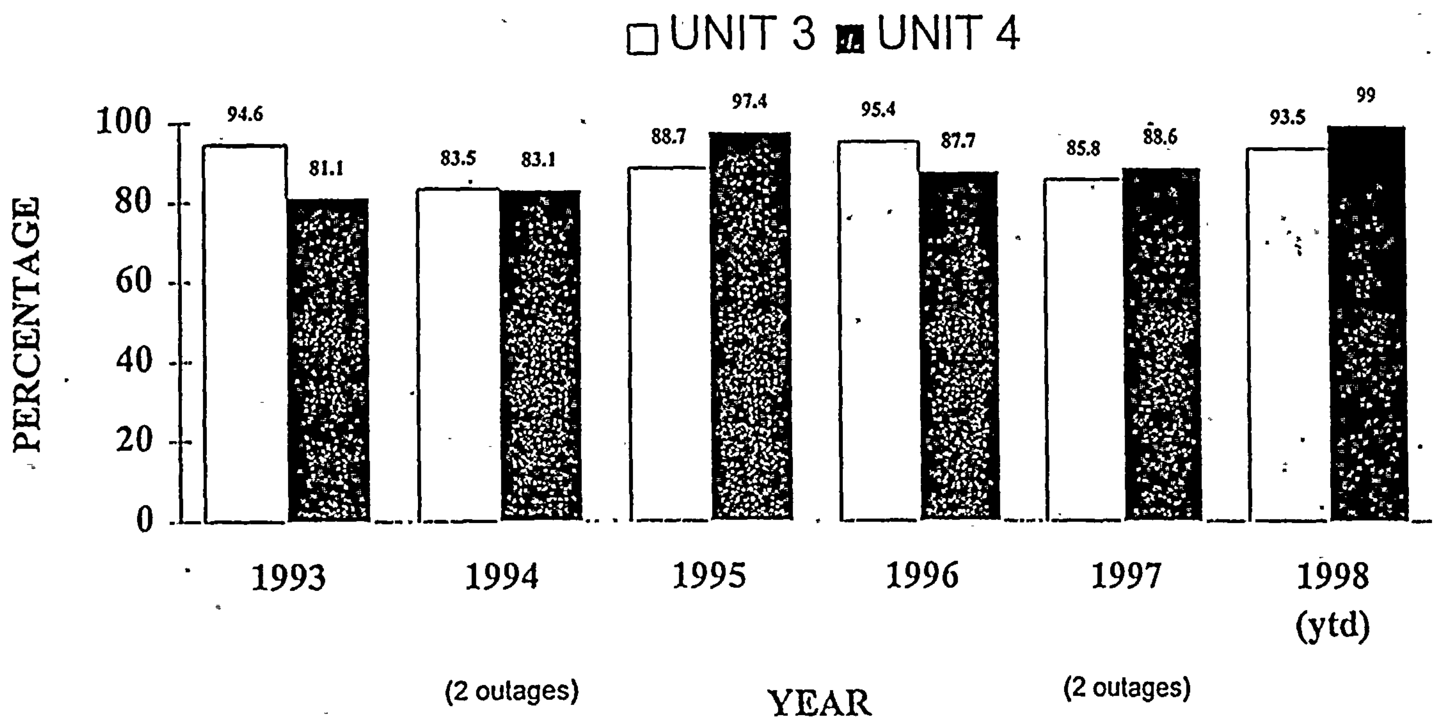
Performance Highlights

- Strong nuclear safety performance
- Efficient refueling outages
- Safe and reliable unit operation
- Continued reductions in station operating cost
- Continued reduction in personnel exposure
- Industrial safety
  - No lost time accidents in 1996/1997
  - Injury rate 0.00
  - Safety 2000 Initiative
  - 2.8 million manhours without a lost time accident

WANO Weighted Overall Performance



Equivalent Availability Factor



1998 FOCUS

- Safe and reliable unit operation
- Continuous improvement
- Develop and maintain a skilled and motivated workforce
- Cost effectiveness

STATION OPERATIONS

D. E. Jernigan

STATION OPERATIONS

Nuclear Safety Culture

- Conservative decision making relative to plant operation
  - Excellent operator response to plant transients
- Strong reactivity management program
- Strong shutdown risk management
  - Full core offload during refueling outages
  - No midloop activities with fuel in vessel
  - No major equipment removed from service until water level >23 feet in reactor cavity
  - Risk Assessment Team chaired by SRO certified engineering supervisor
- Online maintenance program
  - Red Sheet program
- Priority placed on minimizing operator workarounds
- Emphasis on control room "dark board"

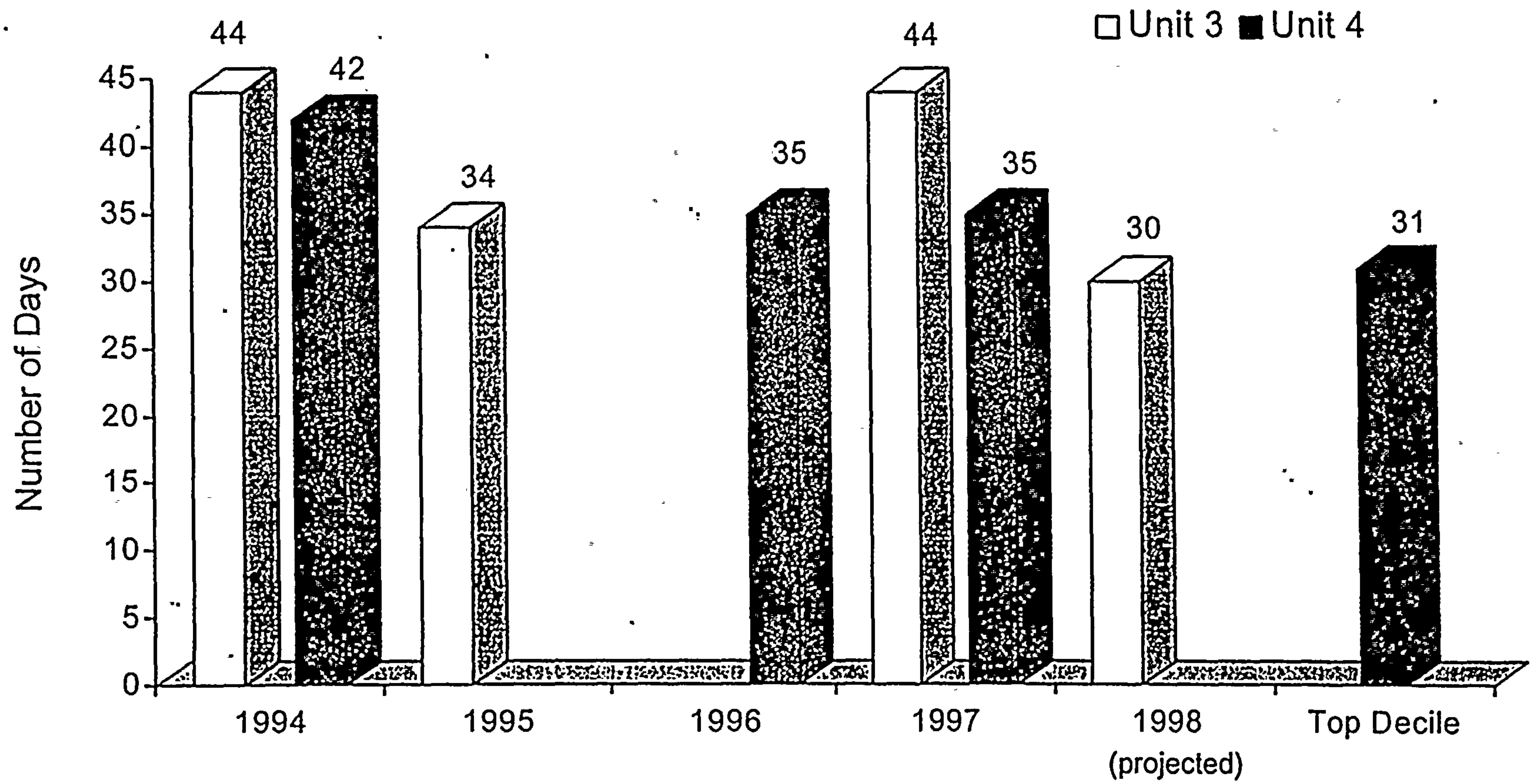
STATION OPERATIONS

Challenges

- Auto Trips during SALP period
  - April, 1997 - Unit 4 - Turbine Control Relay bumped: OTΔT trip
  - July, 1997 - Unit 3 - MSIV closed due to failed seal-in relay
- Station Focus Areas

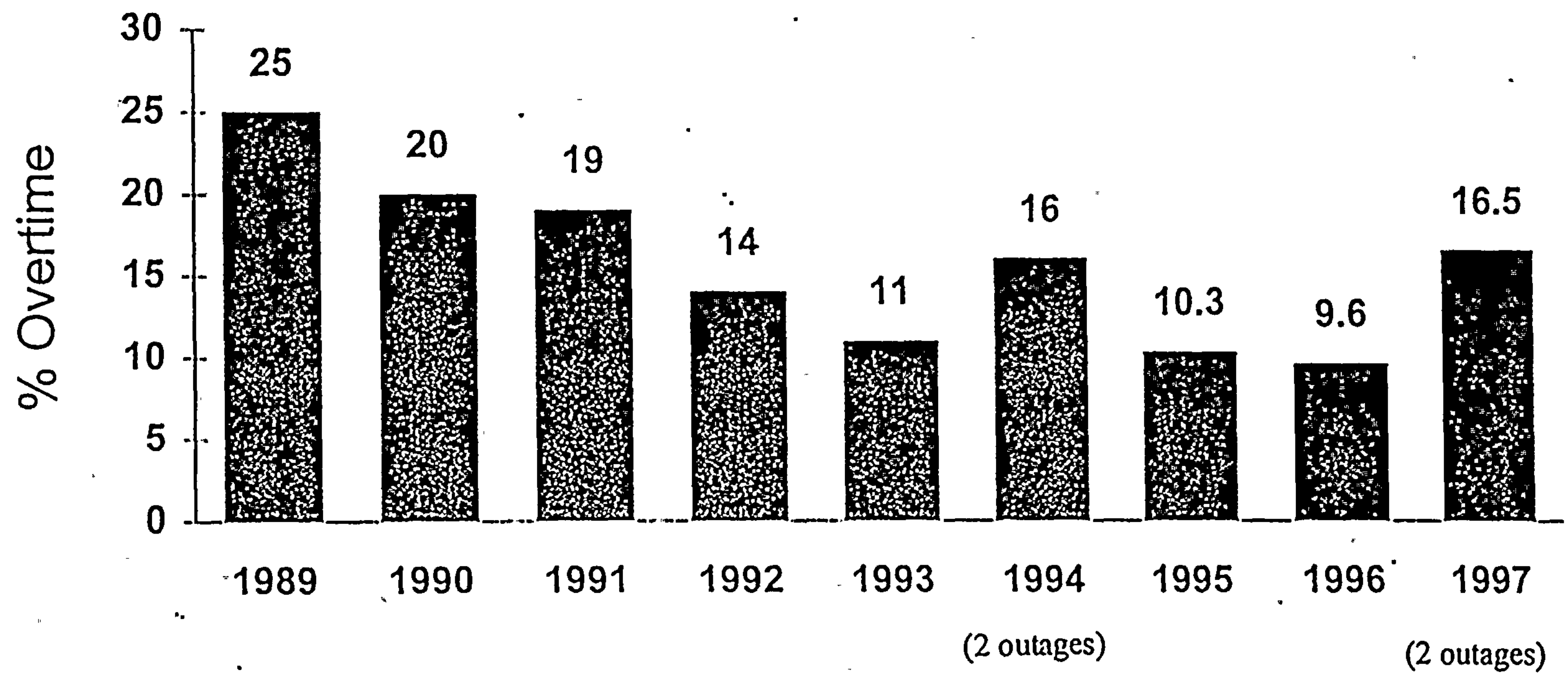
STATION OPERATIONS

Outage Duration



STATION OPERATIONS

Station Overtime



STATION OPERATIONS

Operations Department Self Assessments

- Unit 3 Refueling outage critique
- Unit 4 Refueling outage critique
  - no significant human errors
  - best ever core reload, core physics test, and unit restart
- Performance Improvement International (Dr. Chong Chiu)
  - High performance team training - INPO
  - Human error reduction training - top down
  - Work Control Center reorganization
  - Pre-outage training of critical activities
  - Improved non-licensed operator working conditions
  - Pursuing a 12-hour shift rotation
  - Standardize outage crew briefs (draining the RCS)
- Plant wide self evaluation

Operations Department Self Assessments (cont.)

- Human performance self assessment (1st quarter 1998)
  - Procedure error Condition Reports reduced by 12%
  - Strong conservative decision making and oversight
    - Terminated Unit 4 turbine valve test
    - Load reduction on Unit 3 to tune secondary
  - Effective shift turnover meeting
  - Excellent pre-evolution briefs for load changes and testing
  - "Philosophy of safe reactor operation was supported and reinforced by training and consistent conservative decision making." NIR 97-13
- Simulator self assessment
  - Improvement opportunities identified in crew briefs and communications

STATION OPERATIONS

Station Focus Areas

- Self assessment
- Radiation protection
- Equipment aging
- Valve maintenance
- Employee attrition
- Housekeeping culture
- Efficiency of workforce
- Budget management process

MAINTENANCE

M. O. Pearce

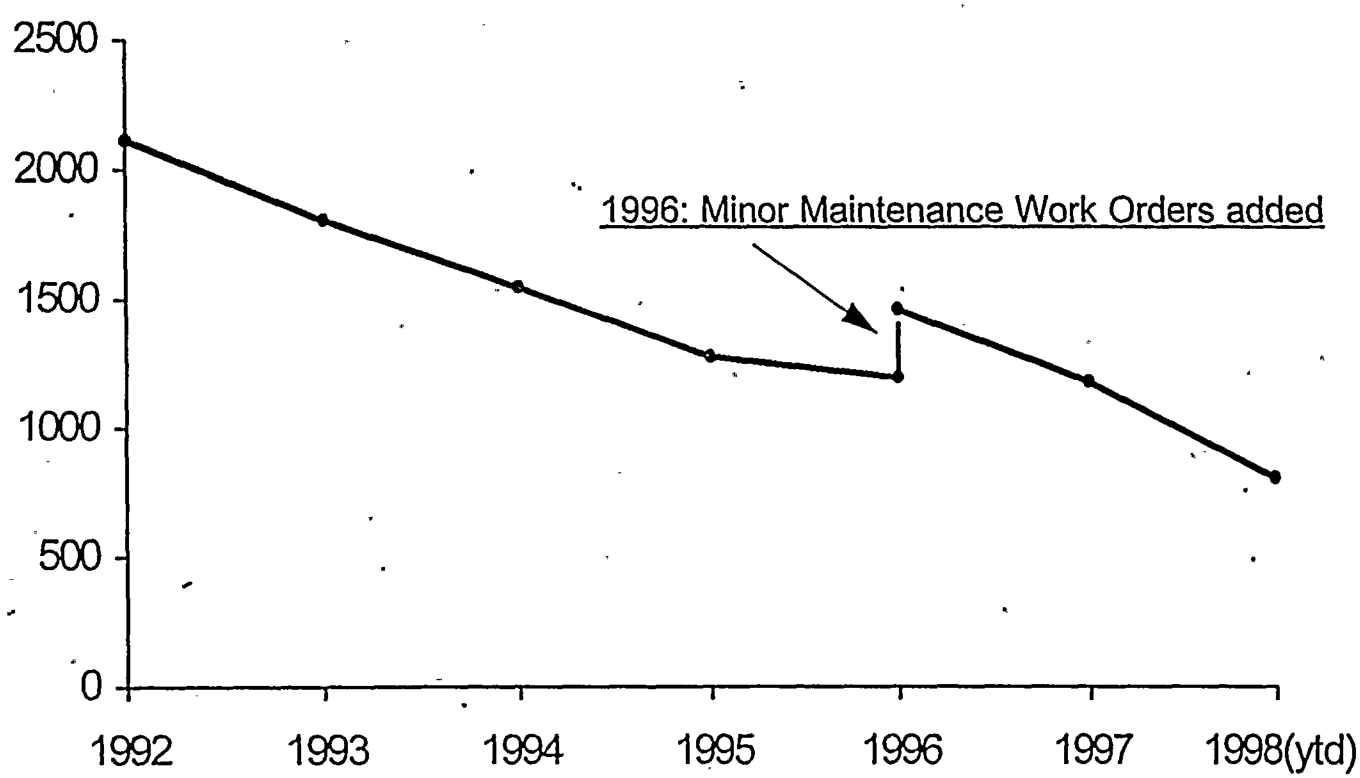
MAINTENANCE

STRENGTHS

- Backlog reduction
  - All work orders
  - Trouble & breakdown work orders
  - Non-outage Control Room work orders & instruments out of service
    - Dark board now routinely achieved
  - Non-outage active oil & water leaks

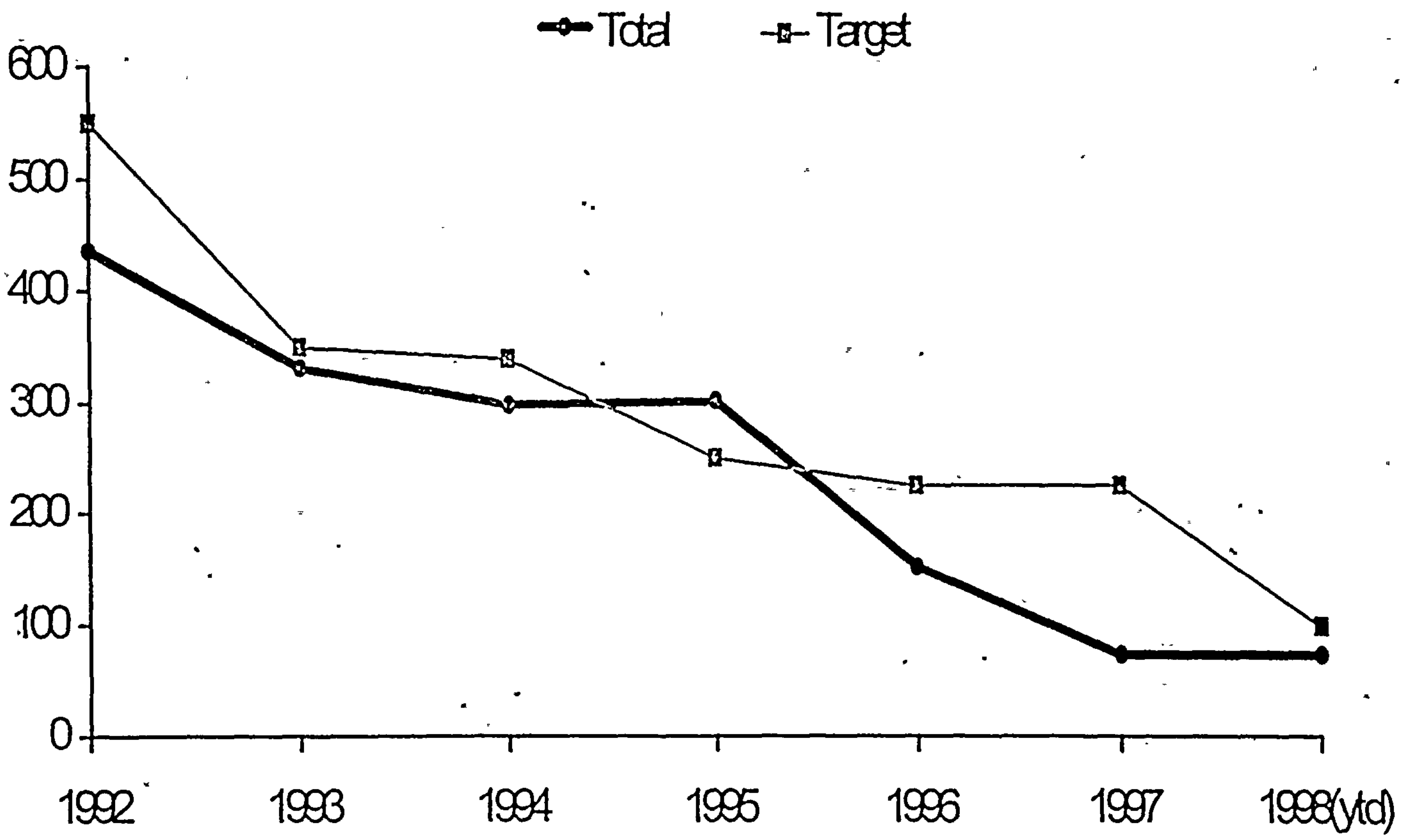
MAINTENANCE

Work Order Backlog
(All Non-Outage, and Trouble & Breakdown Outage)



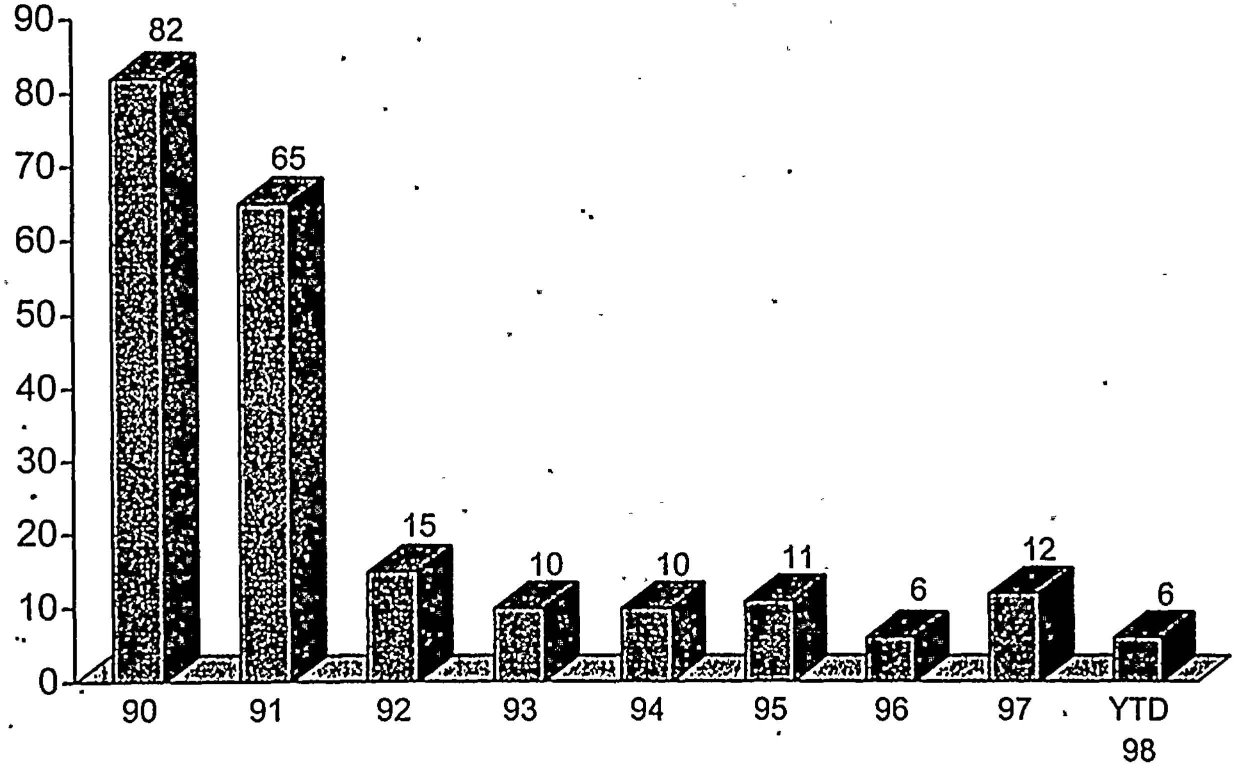
MAINTENANCE

Trouble & Breakdown Work Orders



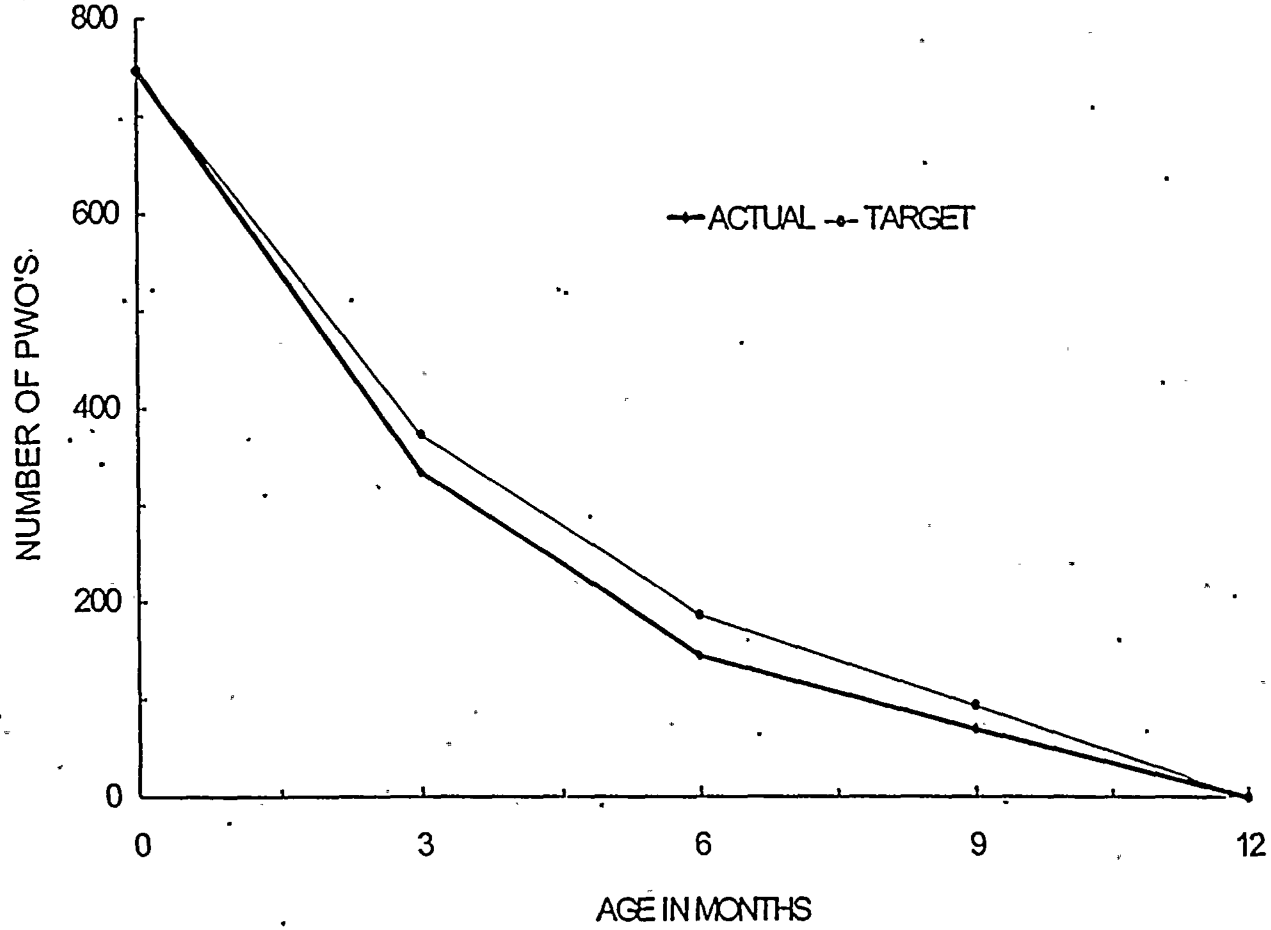
MAINTENANCE

Control Room Deficiency Tags
(Non-Outage)



MAINTENANCE

Non-Outage Work Order Age



MAINTENANCE

STRENGTHS

- 1997 Refueling outages - high availability with short refueling outage performance
  - Unit 3
    - Modifications implemented 42
    - Work orders completed 1954
    - High pressure turbine overhaul
    - Total maintenance craft hours 212,500
  - Unit 4
    - Modifications implemented 20
    - Work orders completed 1547
    - Total maintenance craft hours 192,300

MAINTENANCE

STRENGTHS

- Maintenance Rule integration
  - Risk assessment into on line maintenance

- Strong teamwork, supervisory oversight, and management involvement
  - 4B Reactor Coolant Pump repair
  - Power reduction for turbine valve test, clean and eddy current test the Turbine Plant Cooling Water heat exchangers
  - Strong surveillance program, with best ever Integrated Safeguards Test (Unit 4)

- Strong Maintenance self assessments program

MAINTENANCE

SIGNIFICANT SELF ASSESSMENTS PERFORMED

- Overall internal site evaluation
- Tim Martin & Associates - maintenance & work controls benchmarking analysis; "One of the best performers" in process effectiveness, staffing, practices, & cost
- Overall maintenance assessment (INPO criteria)
- Maintenance Rule
- Preventive maintenance optimization
- Welding program
- Maintenance training
- Refueling outage (2)
- Tool control
- Foreign material exclusion

MAINTENANCE

CHALLENGES

- Control of external corrosion
  - Detection
  - Correction
  - Prevention
- Electronic equipment aging
- Scheduling improvements & workforce efficiency

MAINTENANCE

CHALLENGES ADDRESSED

- Rod control failures
- Hydrogen embrittlement of secondary relief valves
- High head safety injection pump casing leaks
- Material Condition
  - Coatings
  - Housekeeping
  - Foreign material exclusion

ENGINEERING

E. A. Thompson

ENGINEERING

STRENGTHS

- Effective Engineering support of Operations and Maintenance
  - Strong daily plant support
  - Comprehensive outage support
    - Outage risk assessment
    - Equipment overhaul and repair assistance
    - System/task management
  - Maintenance rule implementation

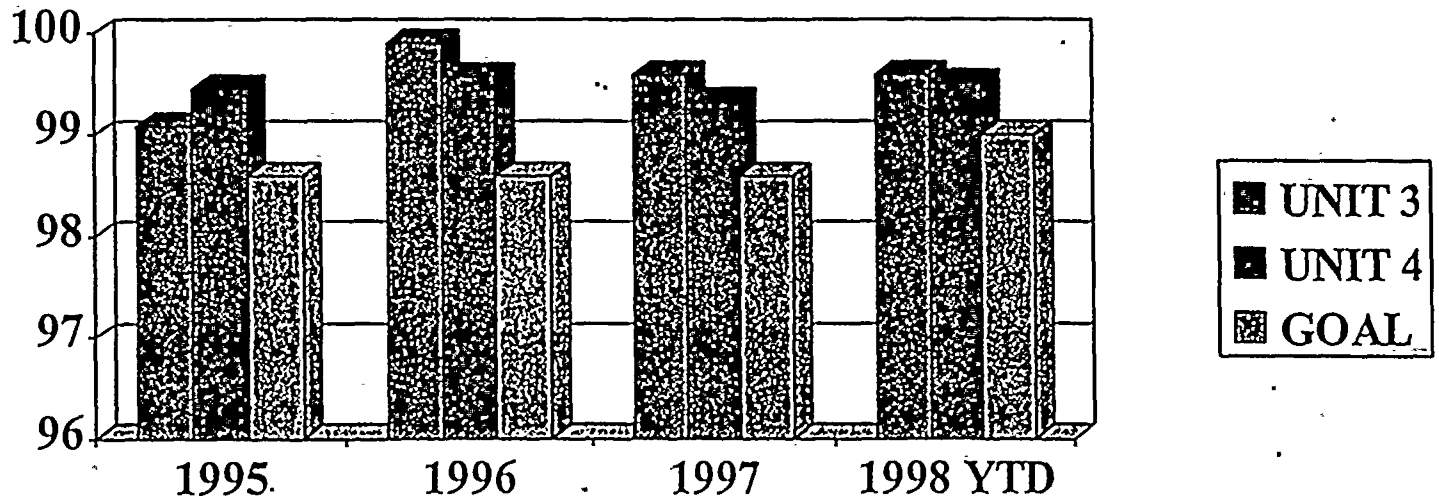
ENGINEERING

STRENGTHS

- Use of Event Response Teams
  - Timely resolution
  - Excellent teamwork
  - Effective corrective actions
- Root cause analysis
  - EDG normal start failures
  - AFW overspeed events
- Assuring excellent equipment and station performance

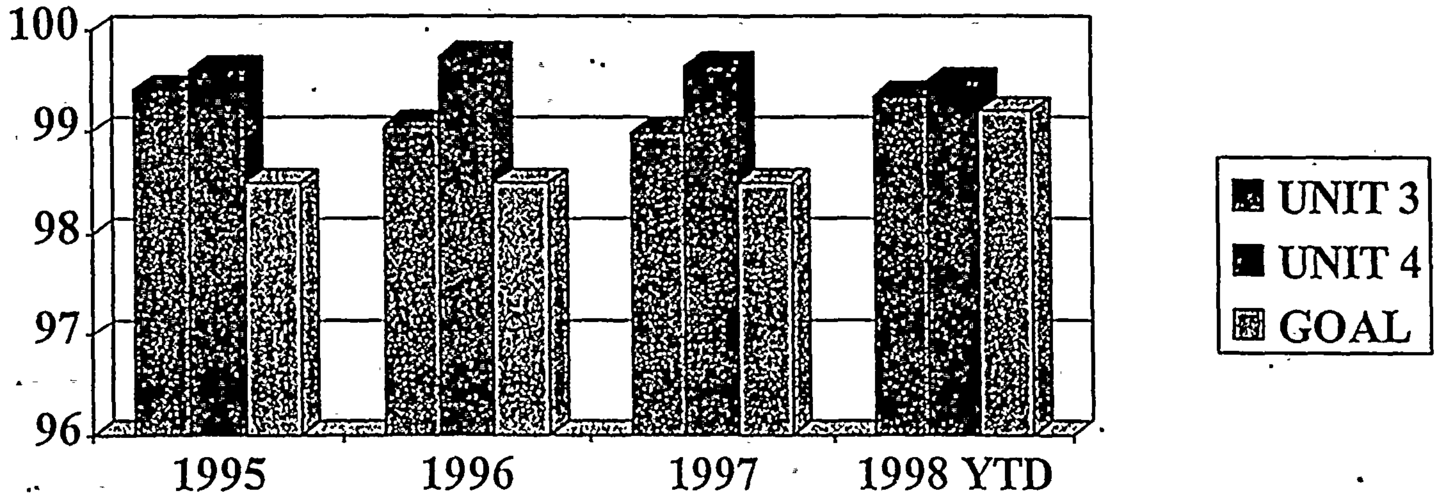
ENGINEERING

HIGH HEAD SAFETY INJECTION AVAILABILITY
(2 YEAR AVERAGE)



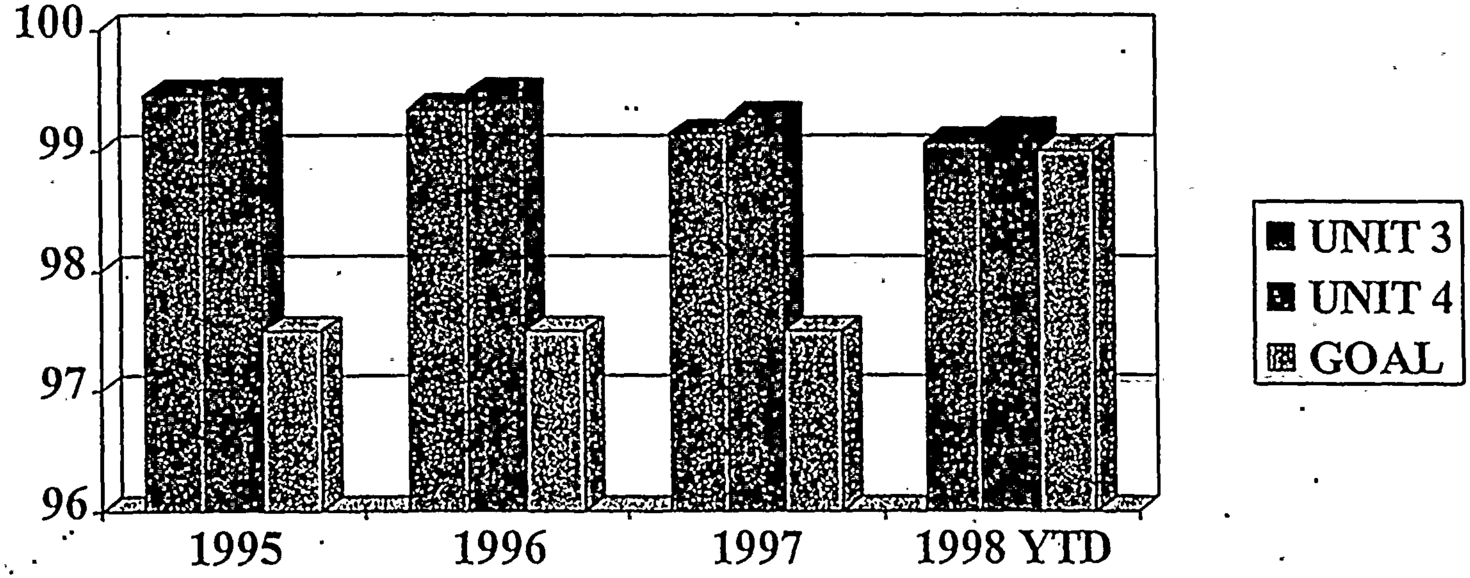
ENGINEERING

EMERGENCY DIESEL GENERATOR AVAILABILITY
(2 YEAR AVERAGE)



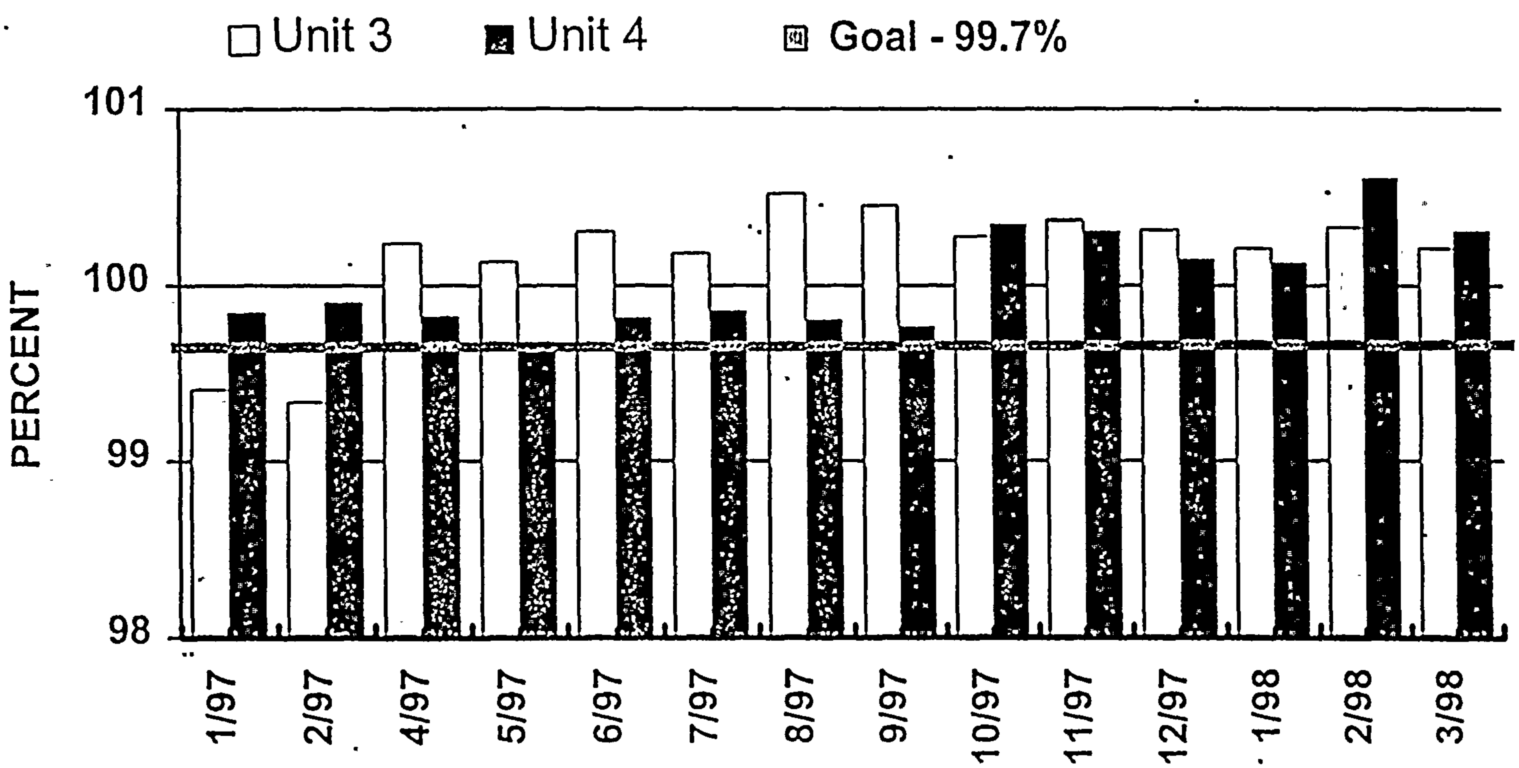
ENGINEERING

AUXILIARY FEEDWATER AVAILABILILTY
(2 YEAR AVERAGE)



ENGINEERING

Thermal Performance



ENGINEERING

STRENGTHS

- REACTOR AND FUELS ENGINEERING
  - No fuel failures for the past three cycles
  - Excellent support and oversight of new fuel and core reload activities
  - World class startup and low power physics testing training, briefing, and coordination
  - Real time support of operations during power maneuvers including written guidance
  - Core design parameter approval by management prior to design
  - Proposed fuel mechanical design changes evaluated via formal engineering evaluation

ENGINEERING

INITIATIVES / UPGRADES

- Training and qualification of new personnel
- Standardization
  - Maintenance rule reports and files
  - System turnover books
  - Equipment trending
  - System walkdown documentation
- External corrosion monitoring
- Electronic equipment aging
- FSAR review project
- Low power physics testing improvements
- License renewal

ENGINEERING

External Corrosion Monitoring

- Recent events
  - Auxiliary feedwater steam supply piping
    - Normally cold, insulated piping
    - Areas where water may accumulate
  - EDG fuel oil supply piping
    - Piping routed in french drain
- Monitoring program enhancements
  - Operations enhanced sensitivity to external corrosion during rounds
    - Crew briefings
    - Continuing training
    - Operator identified EDG fuel oil supply piping

ENGINEERING

External Corrosion Monitoring (cont.)

- Monitoring program enhancements (cont.)
  - System engineering walkdown improvements
    - Revised system engineer walkdown instructions and forms
    - Provided aging assessment field guides
    - Performing on the job training by supervisors and managers
    - Formal walkdown training scheduled to start 7/98
    - Resultdisposition
      * Condition Report, Plant Work Order, minor deficiency list
      * System window reporting with material condition grade
  - Monthly walkdown for safety related and risk significant systems
  - Quarterly walkdown for non-safety and not risk significant systems

ENGINEERING

SELF ASSESSMENT

- Outage engineering support
  - Initiated Design Review Boards for complex modifications
- Condition Report trending
  - Enhanced trending
  - Lower threshold of reporting
  - Fewer anonymous condition reports
- Maintenance rule
- System engineering
- Reactor engineering program and procedures

ENGINEERING

CHALLENGES

- Thermo-Lag upgrades
- Year 2000
- Continued improvement

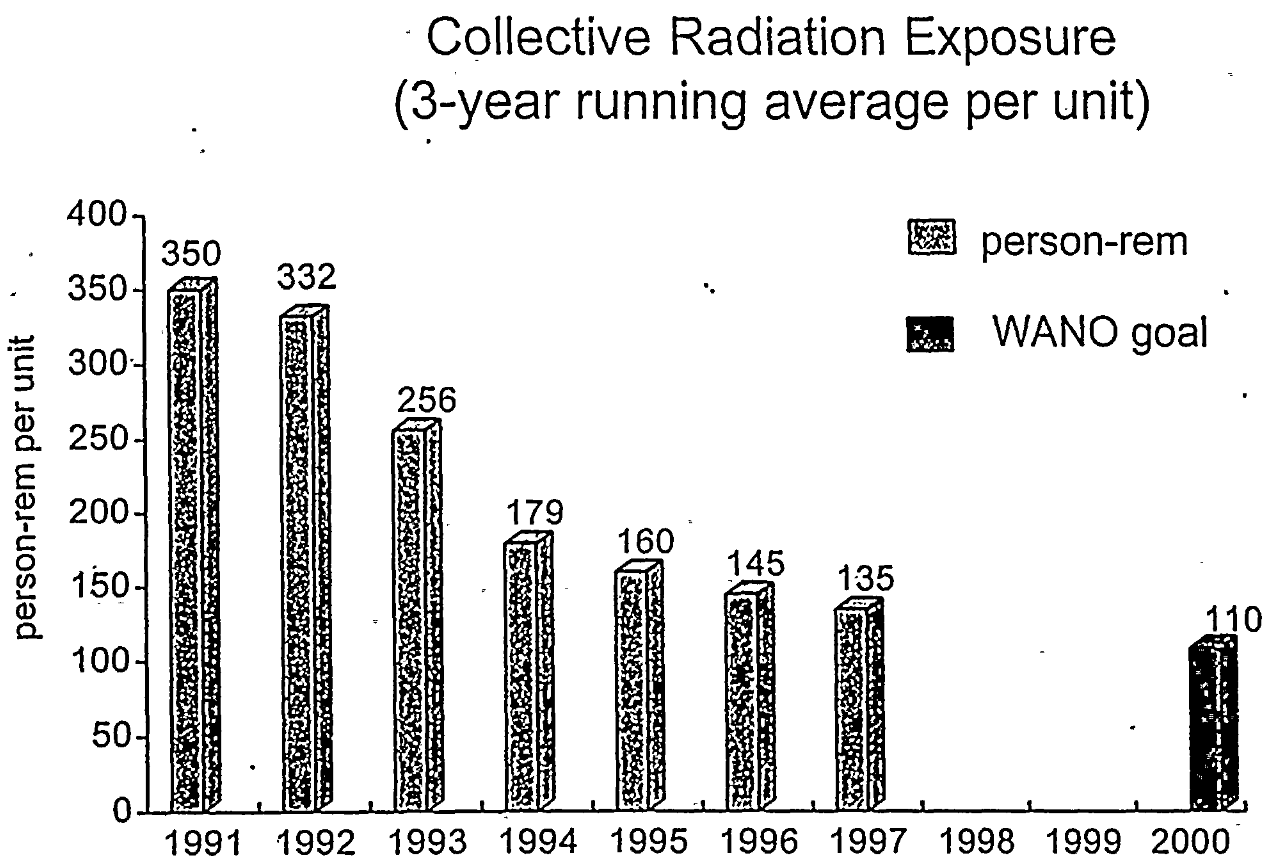
PLANT SUPPORT

HEALTH PHYSICS / CHEMISTRY

J. R. Trejo

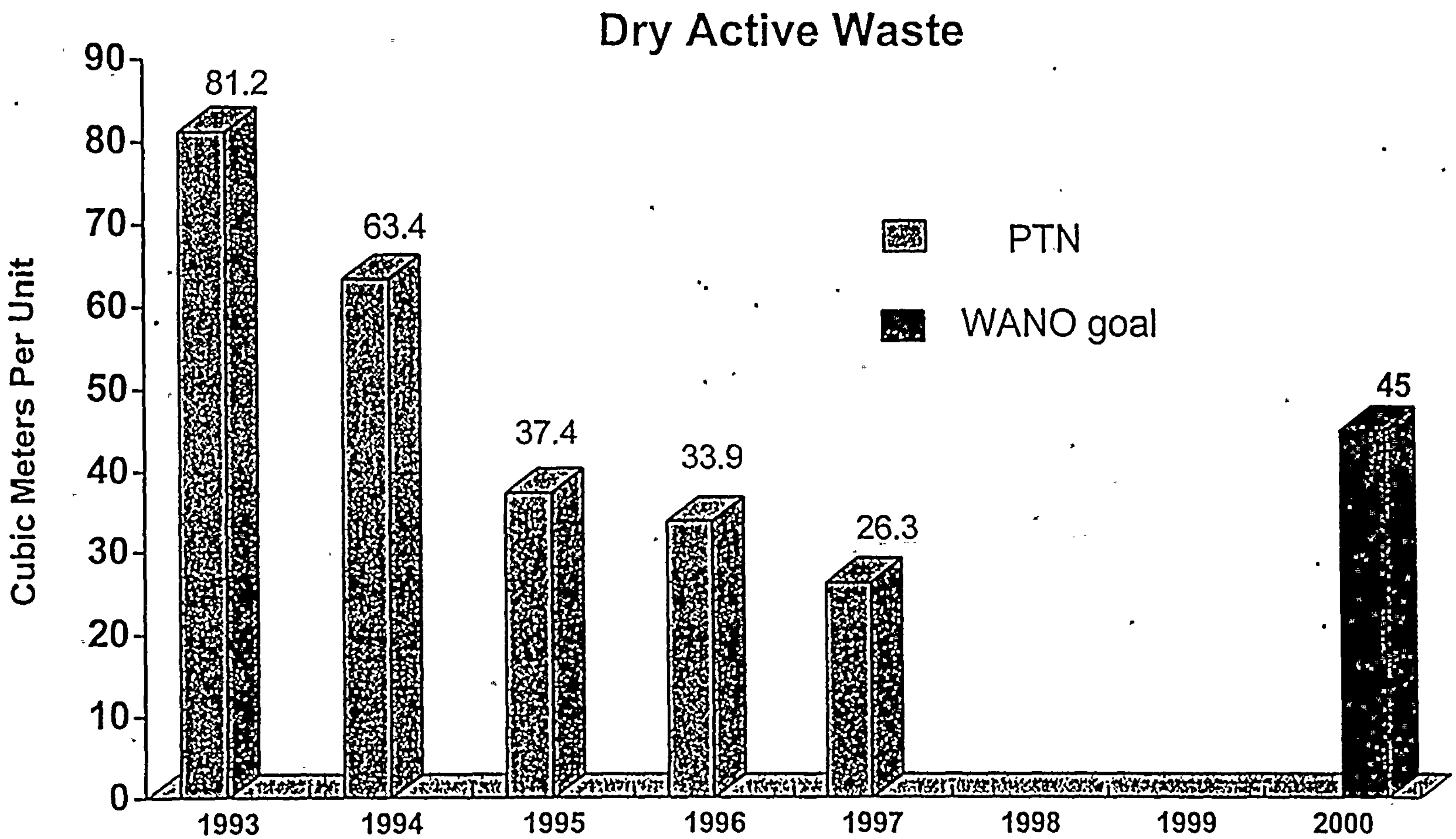
PLANT SUPPORT

Health Physics



PLANT SUPPORT

Health Physics



PLANT SUPPORT

Health Physics

Health Physics Excellence Plan

People Processes Plant

The Objective of this plan is to achieve an exemplary program for radiation protection. The foundation of this plan focuses on a high level of human performance, worker and supervisory involvement, acceptance of radiation protection programs, ownership by radiation protection personnel and radiation workers, and support of operational efficiency and effectiveness.

PLANT SUPPORT

Health Physics

Health Physics Excellence Plan

People

- NRRPT training courses
- Weekly technical training briefs
- Material release training
- Radiation Work Permit knowledge
- Enhanced radiation worker training

PLANT SUPPORT

Health Physics

Health Physics Excellence Plan

Processes

- Recognized strong ALARA program
- Material release procedures strengthened
- Scaler counting for removable radioactivity
- Surveys of certain incoming material
- Conducted industry peer assessment of HP
- HP Control Point staffing increased
- Benchmarking other sites

PLANT SUPPORT

Health Physics

Health Physics Excellence Plan

Plant

- Refurbished main Radiation Controlled Area control point
- New Material Release Building
- Provide improved equipment for free release surveys and personnel monitoring (small article monitors, large area friskers, scalers for smear counting, settings on personnel monitors)

PLANT SUPPORT

Chemistry

	STEAM GENERATOR					
	3 A	3 B	3 C	4 A	4 B	4 C
Preservice Plugs	13	7	19	15	7	9
Previous Outages	4	11	14	1	1	0
Tubes Plugged Last Outage*	3	9	2	0	0	0
Total tubes plugged	20	27	35	16	8	9

Mechanical wear only inservice degradation

100% ECT of all tubes; 3214 tubes per steam generator

100% of all Westinghouse mechanical plugs replaced

Unit 3 Steam Generators in service 2/82 (16 years)

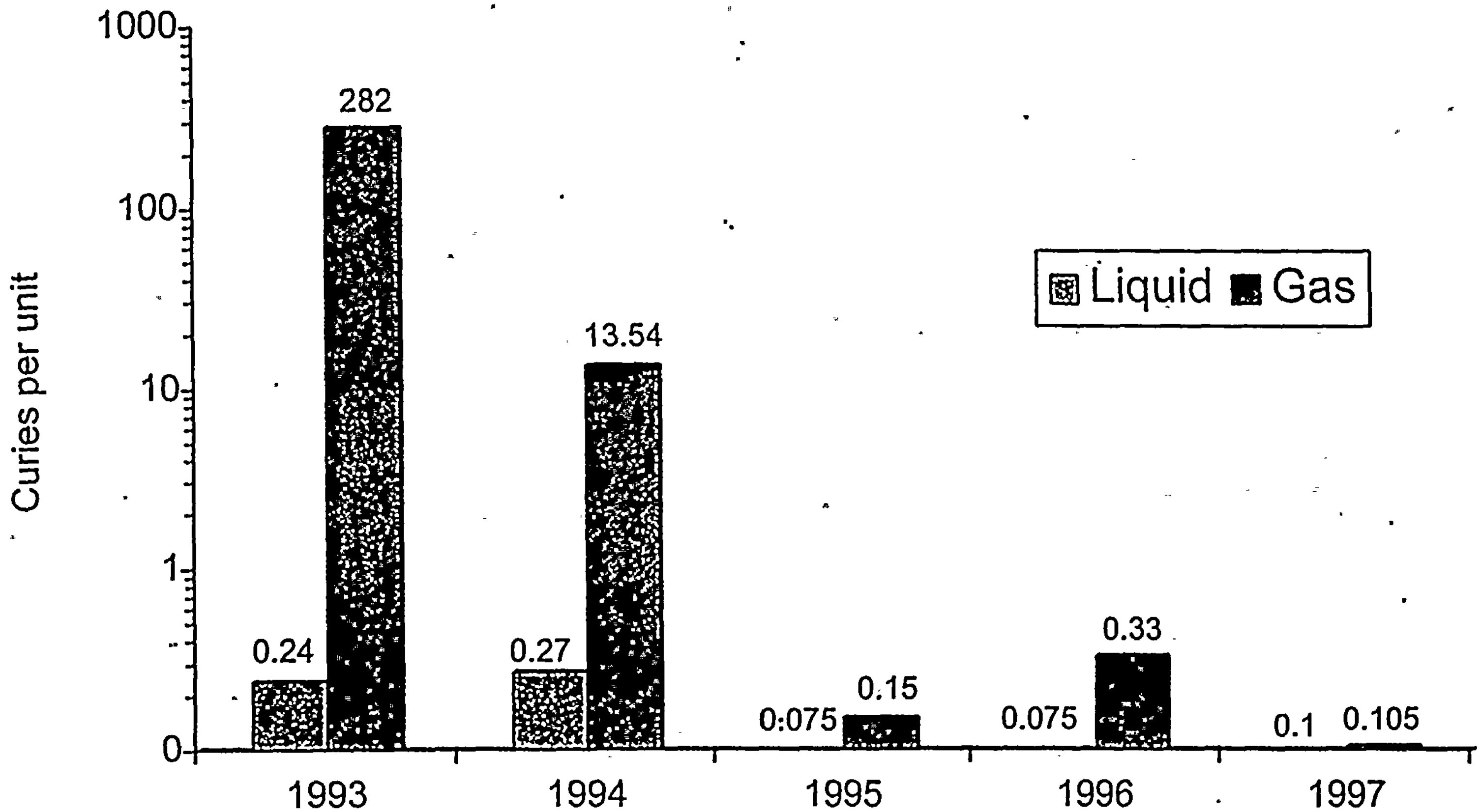
Unit 4 Steam Generators in service 2/83 (15 years)

* 1996; criteria changed to plug on detection

PLANT SUPPORT

Chemistry

Effluent Releases



PLANT SUPPORT

Chemistry

Chemistry Excellence Plan

- Primary chemistry enhancements to reduce dose
- Secondary chemistry enhancements to reduce corrrosion
- Auxiliary cooling water chemistry enhancements to eliminate system degradation
- Continuous improvement

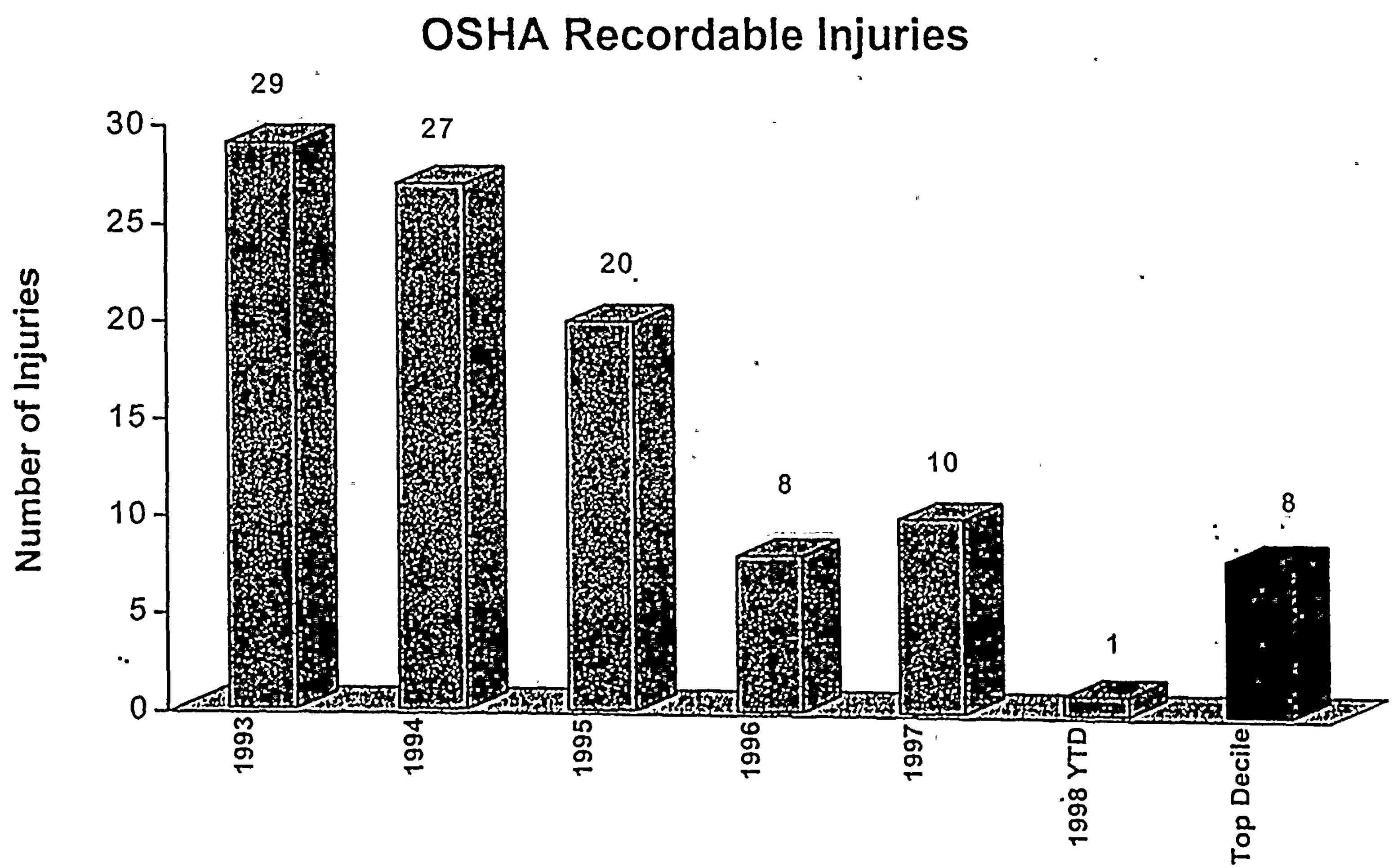
PLANT SUPPORT

Industrial Safety, Emergency Preparedness, Security, Fire Protection

J. E. Kirkpatrick

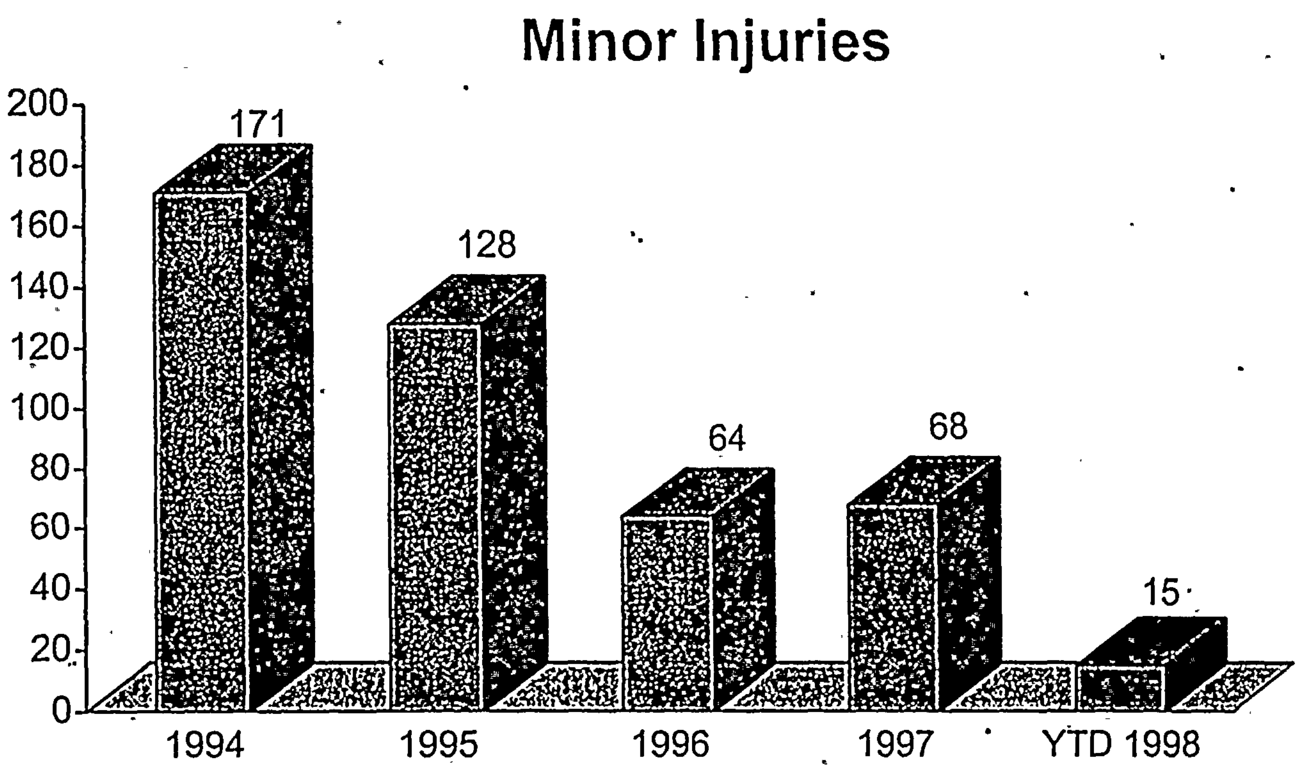
PLANT SUPPORT

Industrial Safety



PLANT SUPPORT

Industrial Safety



PLANT SUPPORT

Emergency Preparedness

STRENGTHS

- Quarterly drill program
- Strong local government relationship
- Strong management support

ACCOMPLISHMENTS

- Successful ingestion exposure pathway exercise, May 1997
- Implemented Severe Accident Management Guidelines, July 1997
- Enhanced pre-hurricane season readiness procedure
- Emergency Operations Facility enhancements
  - staffing, facility, procedures
- Mass casualty mutual aid drill

PLANT SUPPORT

Security

STRENGTHS

- Operational Safeguards Readiness Exercise
  - Upgraded defense on north end of plant
  - Expanded OSRE drill program
  - Relocated armed officers into power block
- FBI mutual aid drill

ACCOMPLISHMENTS

- Vehicle Barrier System final inspection
- Upgraded Firearms Training Simulator
- Self assessment culture
  - Upgraded plans, procedures, and Security Force Instructions
  - Improved plant entrance processes
- Reduced weapons entry attempts
- Reduced loggable events

PLANT SUPPORT

Fire Protection

STRENGTHS

- Fire brigade training (26 state certified members)
- Improved fire protection equipment reliability

ACCOMPLISHMENTS

- Maintained impairments at all time low
- Implemented DETEC system (roving firewatch bar code system)
- Reduced detection system downtime
- Region II/III Fire Protection conference

PLANT SUPPORT

Training

M. L. Lacal

PLANT SUPPORT

Training

STRENGTHS:

- Workforce Development Programs
  - Operations Programs
    - National Nuclear Accrediting Board noted Turkey Point as industry model for operations training programs
    - 5 of 5 SRO candidates received NRC license
  - Maintenance Programs
    - General Maintenance Leader training
    - Valve maintenance training (focus team)
  - Radiation Protection Program
    - Enhanced radiation worker and radioactive material control training (focus team)
  - Engineering Support Program
    - Oral boards administered by licensed Senior Reactor Operator, engineering supervisor, and training instructor
  - Enhanced leadership training
  - Sitewide human error reduction training

PLANT SUPPORT

Training

STRENGTHS

- Effective assessments
  - Continuous assessments of accredited training programs to ensure compliance with NRC Training Rule
  - Comprehensive self-evaluations using industry peers and line supervisors

- Extensive use of mockups / aids
  - Workforce performance improvements
  - ALARA
  - Plant modifications and procedures validated prior to implementation
  - Control Room Simulator used as an effective tool to aid in analysis of significant plant events

PLANT SUPPORT

Training

EXAMPLES OF TRAINING MOCK-UPS / AIDS:

- Plant reference simulator
- Self check (STAR) simulator
- Digital imaging (Digi-Pic)
- PC based version of full-scope simulator
- Valve assembly, disassembly, and seat resurfacing mock-ups
- Flux mapper mock-up
- Lab volt stations for electrical and I&C
- Static inverters and battery chargers
- Various breakers including switchgear breakers, reactor trip breakers
- EDG control panel
- Sequencer panel
- Numerous pumps including charging, S/G feed, RHR
- See-through power plant, containment building mock-up
- Reactor coolant pump seal

Plant Support

Training

STRENGTHS

- Strong and effective line involvement directly contributes to superior station performance
  - Plant Training Advisory Board chaired by Site Vice President
  - Training Review Committee for accredited programs chaired by line department heads
  - Subject Matter Experts effectively used (SROs, Maintenance and Engineering supervisors)
  - Control Room Simulator team training led by operations supervisor

CHALLENGES:

- Continued excellent training programs with given turnovers / new hires
- Development of initial operator license examinations

CLOSING REMARKS

R. J. Hovey