# TURKEY POINT 10 CFR 50.59 REPORT

*FLORIDA POWER & LIGHT COMPANY*
*TURKEY POINT UNITS 3 & 4*

TURKEY POINT PLANT UNITS 3 AND 4

DOCKET NUMBERS 50-250 AND 50-251

CHANGES, TESTS AND EXPERIMENTS

MADE AS ALLOWED BY 10 CFR 50.59

FOR THE PERIOD COVERING

NOVEMBER 14, 1994 THROUGH APRIL 7, 1996

# 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 FSAR update. This report is intended to meet this requirement for the period covering November 14, 1994, through April 7, 1996.

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

TABLE OF CONTENTS

| SECTION 1 | **PLANT CHANGE/MODIFICATIONS** (Continued) | **PAGE** |
|---|---|---|

# SECTION 1

# PLANT CHANGE / MODIFICATIONS

## PLANT CHANGE/MODIFICATION 88-296

UNITS : 3 & 4
TURN OVER DATE : 06/19/95

## PLANT LIGHTING UPGRADE


### Summary:

This Engineering Package replaced and/or supplemented the existing normal plant lighting system in areas were it was found deficient during a systematic walkdown. The objective of this Plant Change/Modification (PC/M) was to improve illumination levels and upgrade the illumination in various areas in order to meet current industry standards. Previous plant lighting intensity measurements were verified, and the quantity and type of lighting fixtures required to upgrade illumination levels in deficient areas were determined. This modification provided minimum illumination levels between 5 and 20 foot candles at the work plane as recommended by the Illumination Engineering Society (IES) Lighting Handbook. This modification did not add or modify security lighting, lighting for fire protection, or emergency lighting powered from the emergency diesel generators (EDGs). In some cases, these modifications required the installation of new lighting fixtures, fixture supports, new raceways, and raceway supports within the plant protected area. These modifications also involved the relocation of existing fixtures and associated supports. Any changes in routing or supports were designed to comply with the seismic criteria in Updated FSAR Appendix 5A to preclude any seismic interactions with safety related structures, systems or components.


### Safety Evaluation:

The normal plant lighting system does not perform any nuclear safety related functions. However, to preclude the possibility or any seismic interactions with safety related structures, systems and components, the mounting and relocation of lighting fixtures and associated structural supports were evaluated in accordance with seismic criteria contained in the Updated FSAR. Based on the evaluation criteria addressed in this PC/M, these lighting 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 89-446

UNIT : 3
TURN OVER DATE : 08/18/95

## REGULATORY GUIDE 1.97
## ENVIRONMENTAL QUALIFICATION (EQ) UPDATE


Summary:

This Engineering Package established the design requirements for modification of the conditioning modules and replacement of three Unit 3 temperature elements (TEs), whose functions are identified in Table 7.5-1 of the Updated FSAR as Regulatory Guide 1.97, Type D, Category 2, variables. These original temperature elements function to monitor: (1) the "A" Residual Heat Removal (RHR) pump discharge temperature; (2) the "B" RHR pump discharge temperature; and (3) the RHR heat exchanger outlet temperature. In order to meet commitments to comply with Regulatory Guide 1.97, the original instrumentation was replaced with instrumentation whose environmental qualifications had been documented for a post-accident environment. These temperature elements do not have any automatic safety functions, but are required to allow operators to monitor RHR system temperatures during the long term post-accident period by displaying their output in the control room.


Safety Evaluation:

The subject temperature elements are not required to perform a safe shutdown function during design basis accidents. They are, however, identified in FSAR Table 7.5-1 as Type D, Category 2 variables which are required to be environmentally qualified. This plant change updates engineering documentation to reflect the addition of these temperature elements to the EQ Program. This modification did not alter the operation of any equipment in the control room or on the Alternate Shutdown panels. Also implementation of these design changes did not require changes to operator actions to cope with off-normal or emergency events. 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. 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 temperature element modifications.

## PLANT CHANGE/MODIFICATION 89-447

UNIT : 4
TURN OVER DATE : 08/25/95

## REGULATORY GUIDE 1.97
## ENVIRONMENTAL QUALIFICATION (EQ) UPDATE

**Summary:**

This Engineering Package established the design requirements for modification of the conditioning modules and replacement of the Unit 3 temperature elements (TEs), whose functions are identified in Table 7.5-2 of the Updated FSAR as Regulatory Guide 1.97, Type D, Category 2, variables. These original temperature elements function to monitor: (1) the "A" Residual Heat Removal (RHR) pump discharge temperature; (2) the "B" RHR pump discharge temperature; and (3) the RHR heat exchanger outlet temperature. In order to meet commitments to comply with Regulatory Guide 1.97, the original instrumentation was replaced with instrumentation whose environmental qualifications had been documented for a post-accident environment. These temperature elements do not have any automatic safety functions, but are required to allow operators to monitor RHR system temperatures during the long term post-accident period by displaying their output in the control room.

**Safety Evaluation:**

The subject temperature elements are not required to perform a safe shutdown function during design basis accidents. They are, however, identified in FSAR Table 7.5-2 as Type D, Category 2 variables which are required to be environmentally qualified. This plant change updates engineering documentation to reflect the addition of these temperature elements to the EQ Program. This modification did not alter the operation of any equipment in the control room or on the Alternate Shutdown panels. Also implementation of these design changes did not require changes to operator actions to cope with off-normal or emergency events. 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. 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 temperature element modifications.

## PLANT CHANGE/MODIFICATION 92-178

UNITS : 3 & 4
TURN OVER DATE : 02/07/95

### INSTALLATION OF SECONDARY SAMPLE LINES TO NEW COLD CHEMISTRY LABORATORY BUILDING

**Summary:**

This Engineering Package upgraded and integrated the existing secondary sampling system with the design of the new Cold Chemistry Laboratory. The overall project included construction of a new ground level laboratory at the southwest corner of the turbine building and associated new sampling tubing to this new laboratory. The secondary sampling system is relied upon by the Chemistry Department to monitor the chemistry of the secondary side steam cycle, which includes the condensate, feedwater, steam, and blowdown subsystems. Accurate and timely testing of the steam cycle at various points can prevent equipment failures or prevent a shortened service life for major plant components. All affected areas of the plant were field inspected to ensure that there would be no potentially adverse interactions with safety systems. All equipment and associated sample tubing were added or modified in phases so that the existing sampling systems remained operational until the new laboratory structure was functional.

**Safety Evaluation:**

The secondary side sampling system does not perform any safety related functions, and it is not addressed by plant Technical Specifications. Tubing, piping, and associated equipment exposed to winds have been designed in accordance the Updated FSAR design requirements for wind loading. These components have also been designed to precluded any interaction with safety related equipment during or following a seismic event. This modification did not change the operation, function or design bases of any structure, system or component important to safety. Consequently, the integration of the existing secondary side sampling system with the design of the new Cold Chemistry Laboratory 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 this modification.

## PLANT CHANGE/MODIFICATION 93-098

UNITS : 3 & 4
TURN OVER DATE : 12/16/94

## CENTRAL RECEIVING FACILITY


### Summary:

This Engineering Package provided the necessary design documents, drawings, specifications, and instructions for reconstruction of the plant's Central Receiving Facility (CRF) following destruction of the original non-safety related building during Hurricane Andrew on August 24, 1992. During this hurricane the CRF was completely destroyed, except for the foundations, slab, internal office walls and other minor appurtenances. The original structure was designed to comply with the South Florida Building Code for hurricane wind speeds up to 120 miles per hour. The new structure was redesigned to provide additional protection against any future hurricane force winds. The new Central Receiving Facility building is a precast concrete structure mounted on a pile supported concrete foundation with nearly the same dimensions and interior facilities throughout the building. The new Central Receiving Facility has been designed for a hurricane wind speed of 150 miles per hour. The CRF will be used for storage of components, materials, and spare parts for plant operations, maintenance and construction activities. This structure functions to provide a centralized location for the receipt, QC/QA inspection, storage and distribution of equipment, components and materials for Nuclear Units 3 and 4.


### Safety Evaluation:

The Central Receiving Facility (CFR) provides a centralized location for the receipt, QC/QA inspection, storage and distribution of equipment, components and materials for the plant. It also provides a location for security inspections of equipment prior to entering the plant protected area. The Central Receiving Facility (CFR) does not perform any safety related functions, quality related functions, or other functions important to safety. The modifications contained in this Engineering Package would not have any adverse effect on plant safety or plant operations. 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 construction of the Central Receiving Facility as documented in this Engineering Package.

## PLANT CHANGE/MODIFICATION 93-099

UNITS : 3 & 4
TURN OVER DATE : 02/14/95

## COLD CHEMISTRY LAB


### Summary:

This Engineering Package provided the necessary engineering specifications, design drawings, instructions, and other documentation for the construction of the Cold Chemistry Laboratory building. The overall project included construction of a new ground level laboratory at the southwest corner of the turbine building, connections to the system sample line tubing, modification of the fire protection piping located below the building foundation, and extension of plant utilities to the laboratory. The secondary sampling system is relied upon by the Chemistry Department to monitor the chemistry of the secondary side steam cycle, which includes the hotwell, condensate, feedwater, steam, and blowdown subsystems. Accurate and timely testing of the steam cycle at various points on the secondary side of the plant can prevent equipment failures or prevent a shortened service life for major plant components. The original secondary side sample points were decentralized and delayed a plant response when timeliness was essential, such as when a condenser tube leak was suspected. The Cold Chemistry Laboratory is a reinforced concrete structure which was designed with sufficient strength to preclude its collapse onto adjacent safety related components. This structure was analyzed and designed for all South Florida Building Code load requirements and for seismic and hurricane/ tornado wind loads in accordance with structural design criteria from the Updated FSAR.


### Safety Evaluation:

The new Cold Chemistry Laboratory and associated secondary sampling system serve no safety function, will not affect any system whose function is to prevent or mitigate the consequences of design basis accidents, and are not addressed by plant Technical Specifications. The Laboratory structure was designed as a Class III structure with sufficient strength to preclude its collapse onto adjacent safety related structures and components. This structure has been designed to preclude any interaction with safety related equipment during a seismic event and meets Updated FSAR design criteria for seismic and wind loadings. Consequently, the construction of the new Cold Chemistry Laboratory to allow secondary side sampling 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 this modification.

# PLANT CHANGE/MODIFICATION 93-108

UNIT : 3
TURN OVER DATE : 02/12/96

## INSTRUMENT AIR SYSTEM COMPRESSOR UPGRADE

### Summary:

This Engineering Package established the design requirements to implement changes to the plant Instrument Air system to increase system capacity while improving its reliability. The Instrument Air system has been operated since 1986 with portable instrument air compressors. Use of the original plant air compressors was discontinued in order to reduce electrical loading on those electrical buses supplied by the emergency diesel generators. As the result of the high operating cost of the portable units, an upgrade of the Instrument Air system was implemented to replace the portable compressors with permanently installed units. This modification implemented an option to delete the original air compressors for both Units 3 and 4, while retaining the electric motor-driven compressor 4S associated with Unit 4. The Unit 3 and 4 instrument air systems retained the original cross-tie between the two systems. The objectives for this design option were to provide one motor-driven compressor for Unit 3 with a capacity capable of meeting normal instrument air demands on both Units 3 and 4, and to provide one diesel-driven compressor on Unit 3 with a capacity equal to the motor-driven unit that would start on low system pressure or loss of voltage to the motor-driven compressor.

### Safety Evaluation:

The plant modifications performed under this Engineering Package do not alter the design bases for the Instrument Air system. The safety related equipment which is supplied by the Instrument Air compressors either fail in their safe positions, have safety related backup nitrogen sources, or have some alternate methods of accomplishing their safety functions. Therefore, the reconfigured compressor arrangement will not affect the ability of this safety equipment to perform its safety function. Consequently, these modifications 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 these Instrument Air system modifications.

## PLANT CHANGE/MODIFICATION 93-109

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| UNIT | : | 4 |
| TURN OVER DATE | : | 11/09/95 |

### INSTRUMENT AIR SYSTEM COMPRESSOR UPGRADE

**Summary:**

This Engineering Package established the design requirements to implement changes to the plant Instrument Air system to increase system capacity while improving its reliability. The Instrument Air system has been operated since 1986 with portable instrument air compressors. Use of the original plant air compressors was discontinued in order to reduce electrical loading on those electrical buses supplied by the emergency diesel generators. As the result of the high operating cost of the portable units, an upgrade of the Instrument Air system was implemented to replace the portable compressors with permanently installed units. This modification implemented an option to delete the original air compressors for both Units 3 and 4, while retaining the electric motor-driven compressor 4S associated with Unit 4. The Unit 3 and 4 instrument air systems retained the original cross-tie between the two systems. The objectives for this design option were to provide one motor-driven compressor for Unit 4 with a capacity capable of meeting normal instrument air demands on both Units 3 and 4, and to provide one diesel-driven compressor on Unit 4 with a capacity equal to the motor-driven unit that would start on low system pressure or loss of voltage to the motor-driven compressor.

**Safety Evaluation:**

The plant modifications performed under this Engineering Package do not alter the design bases for the Instrument Air system. The safety related equipment which is supplied by the Instrument Air compressors either fail in their safe positions, have safety related backup nitrogen sources, or have some alternate methods of accomplishing their safety functions. Therefore, the reconfigured compressor arrangement will not affect the ability of this safety equipment to perform its safety function. Consequently, these modifications 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 these Instrument Air system modifications.

## PLANT CHANGE/MODIFICATION 93-148

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| UNITS | : | 3 & 4 |
| TURN OVER DATE | : | 02/24/95 |

### COLD CHEMISTRY LABORATORY ELECTRICAL TIE-INs

**Summary:**

This Engineering Package provided 480 volt AC power for the new Cold Chemistry Laboratory and provided electrical interfaces into the existing plant telephone, paging, and fire alarm systems. In addition, this modification provided power to the new condenser hotwell sample pumps for both units. The condenser hotwell sample pumps are used to monitor the conductivity of the water in the plant secondary side condensate system. This will enable the plant Chemistry Department to detect condenser tube leakage into the hotwell. The systems installed under this Engineering Package for power, telephone, paging, and fire alarms are not safety related, and the components and associated support structures are Class III. Although not part of this Engineering Package, the overall project included construction of a new ground level laboratory at the southwest corner of the turbine building. The secondary sampling system is relied upon by the Chemistry Department to monitor the chemistry of the secondary side steam cycle, which includes the condensate, feedwater, steam, and blowdown subsystems. Accurate and timely testing of the steam cycle at various points can prevent equipment failures or prevent a shortened service life for major plant components.

**Safety Evaluation:**

The systems installed under this Engineering Package for power, telephone, paging, and fire alarms are not safety related and are not addressed in plant Technical Specifications. The components and associated support structures are Class III. Calculations were prepared to demonstrate that all raceway, supports and duct banks associated with the subject modifications were structurally adequate to preclude seismic interactions and would withstand minimum wind loads per criteria from the Updated FSAR. Although modifications were made to safety related 480 volt Motor Control Centers and fire alarm circuitry, they did not alter the functions, design bases, or operation of any structures or systems important to safety. Consequently, these modifications 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 this modification.

## PLANT CHANGE/MODIFICATION 93-174

UNIT : 3
TURN OVER DATE : 09/28/95

### INSTALLATION OF JIB CRANE IN THE UNIT 3 CONTAINMENT BUILDING AT ELEVATION 58'- 0"

**Summary:**

This modification installed a one ton jib crane at the 58-foot elevation in the Unit 3 Containment Building, along the West side of the equipment hatch. A structural steel beam and plate system was erected to serve as a base for the crane. The crane load capacity was limited to 1500 pounds to avoid lifting components which would require the use of heavy load restrictions. The jib crane was equipped with boom rotating stops and an electric limit switch to prevent striking the containment liner plate with the boom. The jib crane would be used only during operating Modes 5, 6, and defueled due to the potential for load drops which could cause unsafe plant conditions. The hoist was designed to be either removed from containment or left in place during normal plant operation. If left in place this Engineering Package established requirements for securing the hoist. Prior to this modification small loads were normally moved using the main polar crane within containment. During the early and late phases of a refueling outage, large tool boxes, small equipment, and other smaller loads were typically staged or removed from the 58-foot elevation within containment. At these times, however, demands were also placed on the polar crane to handle the normal large loads scheduled for these phases of a refueling outage. Therefore, the jib crane installation was designed to enhance the movement of equipment and tools in and out of containment during refueling outages.

**Safety Evaluation:**

The jib crane was designed using conservative factors of safety, such that the worst loading combinations were compared to normal allowable stresses. The result of the analyses indicated that the containment structure and affected structural steel would retain ample structural margin to carry the loads from the jib crane, even under worst case loading conditions. The jib crane was fabricated from materials that had a minimal impact on containment to ensure that containment integrity and hydrogen analyses would not be impacted. Consequently, this modification did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of this modification.

## PLANT CHANGE/MODIFICATION 94-012

UNIT : 3
TURN OVER DATE : 12/15/95

## CVCS DEMINERALIZER RESIN CHANGE

### Summary:

This Engineering Package evaluated and established criteria for the conversion of all the existing Unit 3 Chemical and Volume Control System (CVCS) letdown demineralizers to a single, economical mixed bed resin configuration. Through integrated use, these mixed bed resins (H-OH) could be employed to perform all the functions of the existing demineralizers. An added benefit of this scheme was that the volume of generated resin wastes could be substantially reduced. Implementation of this modification retained the ability to use the previously approved specialized resins should the need again arise. This Engineering Package included criteria for the redesignation of the existing demineralizers along with associated valves and pressure instrumentation to reflect a new configuration and changes in operating practices. Implementation would require the retagging of demineralizer vessels and associated components, and development of procedural controls on demineralizer use.
Only nuclear grade resins approved for use by the plant chemistry manual would be permitted for use by this package to prevent chemical contamination of the reactor coolant system and fouling of the RCS filters.

### Safety Evaluation:

The chemical characteristics of the demineralizer resins would be monitored and their use administratively controlled to prevent inadvertent boron dilution transients. The boron removal capacity of the demineralizer resins would be bounded by the original CVCS design, and therefore retained the original design capabilities for controlling reactor coolant system activity within acceptable limits. The proposed demineralizer modifications did not create any new interactions with systems that were not already evaluated in the Updated FSAR, and these modifications did not involve any changes in Technical Specifications. Consequently, the demineralizer and resin 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 CVCS modifications.

## PLANT CHANGE/MODIFICATION 94-013

UNIT : 4
TURN OVER DATE : 12/15/95

## CVCS DEMINERALIZER RESIN CHANGE

### Summary:

This Engineering Package evaluated and established criteria for the conversion of all the existing Unit 4 Chemical and Volume Control System (CVCS) letdown demineralizers to a single, economical mixed bed resin configuration. Through integrated use, these mixed bed resins (H-OH) could be employed to perform all the functions of the existing demineralizers. An added benefit of this scheme was that the volume of generated resin wastes could be substantially reduced. Implementation of this modification retained the ability to use the previously approved specialized resins should the need again arise. This Engineering Package included criteria for the redesignation of the existing demineralizers along with associated valves and pressure instrumentation to reflect a new configuration and changes in operating practices. Implementation would require the retagging of demineralizer vessels and associated components, and development of procedural controls on demineralizer use. Only nuclear grade resins approved for use by the plant chemistry manual would be permitted for use by this package to prevent chemical contamination of the reactor coolant system and fouling of the RCS filters.

### Safety Evaluation:

The chemical characteristics of the demineralizer resins would be monitored and their use administratively controlled to prevent inadvertent boron dilution transients. The boron removal capacity of the demineralizer resins would be bounded by the original CVCS design, and therefore retained the original design capabilities for controlling reactor coolant system activity within acceptable limits. The proposed demineralizer modifications did not create any new interactions with systems that were not already evaluated in the Updated FSAR, and these modifications did not involve any changes in Technical Specifications. Consequently, the demineralizer and resin 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 CVCS modifications.

## PLANT CHANGE/MODIFICATION 94-035

UNIT : 3
TURN OVER DATE : 05/01/96

## RTDP RELATED RPS/ESFAS SETPOINT CHANGES

### Summary:

This Engineering Package documents implementation of the Westinghouse Revised Thermal Design Procedure (RTDP) Instrument Uncertainty Methodology (WCAP-13719) to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) for Unit 3. A Plant Licensing Amendment (PLA) was approved by the NRC to allow the RTDP to be incorporated for use in calculating the DNBR limits for Units 3 and 4. The RTDP methodology statistically combines the uncertainties of pressurizer pressure, reactor coolant system (RCS) temperature, reactor power and RCS flow with the Departure From Nucleate Boiling Ratio (DNBR) correlation uncertainty to calculate new DNBR limits. By utilizing the RTDP methodology, increased margin was obtained between the DNBR design and safety analysis limits, which can be utilized to offset thermal uprate and unanticipated DNBR penalties. As a result, the over-temperature (OT$\Delta$T) and overpower (OP$\Delta$T) $\Delta$T trip setpoint, and associated uncertainties were changed. $\Delta$T, $T_{HOT}$, and $T_{COLD}$ instrumentation ranges were also affected. This Engineering Package also changed the setpoint and uncertainty parameters for Steam Generator (S/G) Low and Low-Low Level trip functions as a result of the re-evaluation of the process measurement parameter. These changes provided a more accurate assessment of the PMA term and allow for additional operating margin to reduce the possibility of spurious reactor trips during low power transients. The setpoint related changes were not a direct result of RTDP, but were included in the accident analysis evaluations to support application of the RTDP methodology. Another change involving a time delay for the reactor coolant pump undervoltage trip was also addressed by this package.

### Safety Evaluation:

These RPS/ESFAS setpoint modifications did not physically alter any equipment important to safety. This modification did not change the operation, function or design bases of any structure, system or component important to safety as described in the UFSAR and no new hazards were created that could cause an accident different from those previously analyzed. Consequently, these RPS/ESFAS setpoint changes did not involve an unreviewed safety question. Due to the changes in RPS/ESFAS setpoints, NRC approval was obtained for the Technical Specification changes in advance of Engineering Package implementation. Therefore, specific NRC approval was not required for implementation of the Engineering Package modifications.

# PLANT CHANGE/MODIFICATION 94-045

UNIT : 3
TURN OVER DATE : 05/15/95

## REPLACEMENT OF CONTAINMENT CATHODIC PROTECTION SYSTEM

### Summary:

This Engineering Package was developed to allow replacement of the cathodic protection system for the Unit 3 containment structure. Various parts of the containment cathodic protection system had failed over the past several years so that current flow from the anodes had significantly decreased and was not functioning properly. The suspected cause of the problem was corrosion of the wire connections at the anodes and electrodes and the design of the original system did not provide for replacement or maintenance of the system anodes or electrodes. This Engineering Package installed an anode bed, concrete manhole, ductbank, drain pipe, rectifier, safety disconnect switch, anode junction box, reference electrode test stations, and associated conduit and cable. This replacement cathodic protection system used a deep ground anode bed drilled beneath the containment. This anode system was designed for a 20-year life and had provisions for the future replacement of its components, if necessary. The cathodic protection system is one of several means by which the reactor containment structure is protected from corrosion. Other means include: (1) a corrosion preventing grease, pumped into the tendon sheathing and caps, to protect the containment building post-tensioning system; (2) coatings of inorganic primer and a finish coat applied to exposed surfaces of the containment liner plate, except for the floor; and (3) a surveillance program for tendons, used to monitor containment structural integrity and corrosion.

### Safety Evaluation:

The function of the containment cathodic protection system is to protect the containment liner plate, reinforcing steel, and tendon assemblies from the long-term effects of corrosion; this system is only one of several means used to protect the containment structure from corrosion. Based on the above, the cathodic protection system performs no nuclear safety related functions. The installation and final configuration of this modification did not create the possibility of any adverse interactions with existing safety related structures, systems, or components. Consequently, the modification in this Engineering Package did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 94-046

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| UNIT | : | 4 |
| TURN OVER DATE | : | 05/05/95 |

### REPLACEMENT OF CONTAINMENT CATHODIC PROTECTION SYSTEM


**Summary:**

This Engineering Package was developed to allow replacement of the cathodic protection system for the Unit 4 containment structure. Various parts of the containment cathodic protection system had failed over the past several years so that current flow from the anodes had significantly decreased and was not functioning properly. The suspected cause of the problem was corrosion of the wire connections at the anodes and electrodes and the design of the original system did not provide for replacement or maintenance of the system anodes or electrodes. This Engineering Package installed an anode bed, concrete manhole, ductbank, drain pipe, rectifier, safety disconnect switch, anode junction box, reference electrode test stations, and associated conduit and cable. This replacement cathodic protection system used a deep ground anode bed drilled beneath the containment. This anode system was designed for a 20-year life and had provisions for the future replacement of its components, if necessary. The cathodic protection system is one of several means by which the reactor containment structure is protected from corrosion. Other means include: (1) a corrosion preventing grease, pumped into the tendon sheathing and caps, to protect the containment building post-tensioning system; (2) coatings of inorganic primer and a finish coat applied to exposed surfaces of the containment liner plate, except for the floor; and (3) a surveillance program for tendons, used to monitor containment structural integrity and corrosion.


**Safety Evaluation:**

The function of the containment cathodic protection system is to protect the containment liner plate, reinforcing steel, and tendon assemblies from the long-term effects of corrosion; this system is only one of several means used to protect the containment structure from corrosion. Based on the above, the cathodic protection system performs no nuclear safety related functions. The installation and final configuration of this modification did not create the possibility of any adverse interactions with existing safety related structures, systems, or components. Consequently, the modification in this Engineering Package did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 94-059

UNITS : 3 & 4
TURN OVER DATE : 08/25/95

## DIESEL ENGINE DRIVER FOR STANDBY STEAM GENERATOR FEEDWATER PUMP P82B

### Summary:

This Engineering Package involved replacing the electric motor driver of the "B" Standby Steam Generator Feedwater Pump (SSGFP) with a diesel engine driver and its associated power supply and controls. The existing motor driven SSGF pumps were powered from the nonsafety 4160 volt C-Bus. In the event of a loss of offsite power these pumps could be powered by the nonsafety Cranking Diesels. The diesel driven pump, therefore, eliminated the sole dependence on the nonsafety related Cranking Diesel Generators for power. The function of the Standby Steam Generator Feedwater Pumps is to provide a source of make-up water to the Steam Generators during plant startup, shutdown, and for hot standby. These pumps function during these plant evolutions and obviate the need for the safety related Auxiliary Feedwater System (AFW) to perform these normal operating functions. Credit for use of the SSGFPs is also considered in the Appendix R Safe Shutdown Analysis for the plant. In the event of a postulated fire, which renders both trains of AFW unavailable, the SSGF pumps would be used for steam generator makeup. Besides, safe shutdown analysis requirements, the SSGFPs have Technical Specification operability requirements which govern their availability. In advance of implementation of these modifications, FPL received approval from the NRC for the replacement of the B pump motor with a diesel engine driver. The NRC's safety evaluation approved the electric motor replacement on the B pump and removed all requirements for the Cranking Diesel Generators.

### Safety Evaluation:

The modification of the Standby Steam Generator Feedwater pump and associated Technical Specification changes were approved by the NRC in license amendments. Accordingly, use of the diesel engine driven SSGFP to eliminate sole dependence on the Cranking Diesel Generators had received NRC approval. The diesel engine driven pump will have performance characteristics comparable to the original electric motor driven pump, and this modification did not have an adverse affect on plant safety functions. Consequently, the SSGFP modifications did not involve an unreviewed safety question. Since the SSGFP Technical Specifications were approved in advance of Engineering Package implementation, specific NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 94-083

UNITS : 3 & 4
TURN OVER DATE : 02/08/95

## CONTROL ROOM MODIFICATIONS

### Summary:

This Engineering Package (EP) provided for the modification of certain control room features. Installation of operations workstations and other structural changes, and relocation and upgrade of communications systems were implemented to enhance the working environment of the control room. The existing observation booth, Nuclear Plant Supervisor (NPS) desk, Digital Data Processing System (DDPS) printers and cabinets were removed and workstations for the Assistant Nuclear Plant Supervisor (ANPS) were added. Existing phones, PA communications, and the PB control station box on the NPS desk was relocated. Primary notification phones, ENS, NAWAS, and HOT RING DOWN phones were relocated to the Nuclear Watch Engineers office (NWE). Also the Reactor Control Operator (RCO) desks were replaced to accommodate personal computers. Similar modifications would also be performed in the Simulator Training Building. The control room provides a habitable area for plant operations during normal and accident conditions. The material used inside the control room was selected for its low combustible values, flame spread properties and smoke development indices. The workstations provided in the control room were designed to avoid any obstructions to the range of view or movement of control room personnel. The communication systems provide a reliable means of communication inside the plant and with outside agencies during any plant condition. The communication components were designed for accessibility and adequacy of mounting.

### Safety Evaluation:

The observation booth, ANPS workstations, RCO desks and communication systems do not perform any safety related functions. However, they are located inside the control room in the vicinity of safety related systems. These modifications did not affect any safety related system or component, and did not affect any Technical Specification requirements. Consequently, the safety evaluation contained in this engineering package demonstrated that these modifications did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these control room modifications.

## PLANT CHANGE/MODIFICATION 94-090

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|---|---|---|
| UNITS | : | 3 & 4 |
| TURN OVER DATE | : | 06/07/95 |

## REPOWERING OF RADIATION MONITORS


**Summary:**

This Engineering Package repowered the Unit 3 Spent Fuel Pit Vent Stack Radiation Monitor RAD-3-6418 and the Common Plant Vent Stack Radiation Monitor RAD-6304 to reduce the plant's dependence on the five nonsafety grade cranking diesel generators. The monitors were repowered from a safety related power source. These radiation monitors are classified as Regulatory Guide 1.97 post-accident monitoring Category 2, Type C instrumentation and must be supplied from a highly reliable power source. The existing power source was a non-vital power supply. Updated FSAR Section 7.5 stated that cranking diesel generators were relied on as a power source for the above radiation monitors in order to meet the Category 2 power source requirement. In order to better meet these requirements the monitors were repowered from a safety related power source, which is automatically backed by a safety related emergency diesel generator. The new 120 volt AC power supply was taken from a panel associated with vital Motor Control Center 4D. The new power supply cables were routed in newly installed conduits and existing cable trays. During the period in time when the Unit 3 Spent Fuel Pit Vent Stack Radiation Monitor RAD-3-6418 was out of service, fuel movement in the spent fuel pool was suspended. The power panel which powers the two radiation monitors was to be taken out of service during a Unit 4 shutdown when fuel movement in the Unit 3 spent fuel pool was also unlikely.


**Safety Evaluation:**

This evaluation examined the potential for newly installed conduit to interact with existing safety systems. The conduits within the auxiliary building were designed with seismically analyzed supports, and outdoor conduit runs were designed for applicable wind and seismic loadings as per Updated FSAR requirements. The additional loads for cable trays were found to be adequately supported. Consequently, the new power supply cables had no adverse effect on existing safety related structures or systems, and 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 94-104

UNITS : 3 & 4
TURN OVER DATE : 12/22/95

### PERMANENT INSTALLATION OF SERVICE AIR COMPRESSORS AND ASSOCIATED EQUIPMENT


**Summary:**

This Engineering Package documented the system design for the permanent installation of the service air compressors and associated equipment to support Temporary System Alteration (TSA) 3-93-13-21. This TSA replaced the temporary rental air compressors with permanent air compressors and associated equipment. This Engineering Package permanently relocated and installed the above ground piping, drains and valves, buried piping, piping supports and electrical power supply to the compressors. Permanent local air flow indication was also added. The service air system provides a reliable source of air for pneumatic tools and miscellaneous plant nonsafety equipment used during normal plant operation, refueling outages, and for maintenance activities. The service air system provides air to air hose stations throughout the plant at all elevations. Prior to Hurricane Andrew on August 24, 1992, service air use for Units 3 and 4 was supplied from the two fossil plants located adjacent to Units 3 and 4. After the hurricane, several temporary rental air compressor units were located on site to meet the plant's service air demands.


**Safety Evaluation:**

The nonsafety related service air system serves no safety function, since it is not required to mitigate the consequences of any design basis accidents analyzed in the Updated FSAR. However, the piping and instrument installed with this package in the turbine building are seismically supported to preclude any adverse interactions with safety related structures, systems or components in compliance with design basis requirements. These modifications did not have an adverse effect on plant safety or plant operations. Consequently, these modifications 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 these modifications.

# PLANT CHANGE/MODIFICATION 94-111

UNIT : 3
TURN OVER DATE : 10/06/95

## ROD CONTROL CRDM TIMING CHANGES

### Summary:

This Engineering Package was developed to implement Control Rod Drive Mechanism (CRDM) current order timing changes and additional rod control system surveillances as recommended by Westinghouse in their Technical Bulletin NSD-TB-94-05-R0, which was issued to address the Salem Plant Rod Control System event on May 27, 1993. During this event at the Salem Plant, the Rod Cluster Control Assemblies (RCCA) responded inconsistently to corrupted current orders, such that one control rod moved out while inward motion was demanded by the control system. Responding to this event, the NRC issued Generic letter (GL) 93-04 requesting a response to the events at Salem. FPL responded to GL 93-04 by performing a nuclear fuels analysis that demonstrated that existing DNB margins in the Unit 3 and 4 fuel designs were adequate to prevent exceeding fuel design limits on an asymmetrical rod withdrawal event, such as had occurred at Salem. The modifications addressed in this Engineering Package consisted of changing diode connections on each slave cycler decoder printed circuit board located in the Rod Control system logic cabinet. Additional surveillances of the Control Rod System were also implemented to detect single failures which do not affect rod motion, and therefore, are not detectable by routine rod insertion/withdrawal.

### Safety Evaluation:

The Rod Control system is not required to function to achieve or maintain safe shutdown or mitigate the consequences of design basis accidents. However, this system is important to safety, since it controls reactor power during normal operation, and its failure can initiate design basis accidents documented in the Updated FSAR. This modification ensures that a Salem type failure will cause the affected rod control group/bank to insert rather than withdraw, which has been shown to be bounded by the consequences of other design basis accident events already analyzed in the Updated FSAR. Therefore, with the revised CRDM timing modifications, the current licensing basis analyses remain bounding and 1967 Proposed GDC 31 continue to be met for all single failures within the control rod system. Consequently, these CRDM 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 this modification.

## PLANT CHANGE/MODIFICATION 94-119

UNITS : 3 & 4
TURN OVER DATE : 06/08/95

## DESIGN OF PLANT UTILITY EXTENSIONS TO THE CAFETERIA


### Summary:

This Engineering Package established design requirements for the construction of plant utility extensions to a new cafeteria facility that was installed for use by onsite plant personnel. The cafeteria facility is a single story, free standing structure located immediately south of the nuclear administration building and occupying the site of the old construction craft building. This location is outside of the power block just inside the south periphery of the site protected area. The modifications included the utility connections to the plant fire protection, potable water, storm sewer, sanitary sewer, communications, plant paging, and electrical power systems. The cafeteria structure and associated utilities perform no safety related functions and the connections to existing plant systems were either nonsafety related, or nonsafety related portions of existing plant systems.


### Safety Evaluation:

The cafeteria structure and associated utilities perform no safety related functions and the connections to existing plant systems were nonsafety related. No safety system design functions or design bases were adversely impacted by the Engineering Package utilities or associated connections to existing plant utilities. The effects of the new cafeteria and its utilities on the fire protection and security features of the plant were examined, and no adverse effects on fire protection or security requirements were identified. Consequently, the utility modifications addressed under this Engineering Package did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 94-122

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| UNITS | : | 3 & 4 |
| TURN OVER DATE | : | 01/19/95 |

## REPLACEMENT OF CONTROL ROOM CEILING PANELS


**Summary:**

This Engineering Package provided the necessary design requirements for replacement of the ceiling panels in the Units 3 and 4 control rooms. In order to maintain a visual similarity to the ceiling in the control room, the ceiling panels in the plant simulator were also replaced with the same type as used for the control room. This Engineering Package was issued as part of FPL's resolution for NRC Unresolved Safety Issue (USI) A-46 (Generic Letter 87-02) and is in response to a concern raised by the Seismic Review Team (SRT) during the USI A-46 walkdown at the plant. Based on seismic experience data, the SRT identified the existing aluminum "eggcrate" panels as of a type used for the ceiling panels in the control room, and these panels represented a potential personnel hazard to control room operators. The sharp edges of these panels had the potential to injure control room operators should the panels become dislodged and fall during a seismic event.


**Safety Evaluation:**

The control room ceiling panels are a passive architectural feature that do not have any safety functions. However, the design and installation of the ceiling panel modifications was evaluated against fire protection and human factors criteria to ensure that they would not affect licensed conditions for the control room. The modifications documented in this Engineering Package do not have an adverse effect on plant safety or plant operations. 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 ceiling panel modifications.

# PLANT CHANGE/MODIFICATION 94-135

UNIT : 3
TURN OVER DATE : 09/19/95

## INSPECTION, REPAIR AND MODIFICATION OF THE UNIT 3 INTAKE STRUCTURE

### Summary:

This engineering package was prepared to restore the Unit 3 Intake Structure slab, supporting Screen Wash Pumps 3P14 and P14, to its original or equivalent design configuration, thereby ensuring adequate long-term performance of the structure. Concrete walls for the Circulating Water Pump (CWP) 3B2 bay were inspected for any active corrosion of the steel reinforcement within the walls. The concrete slab for the CWP 3B2 bay, which supports the two Screen Wash pumps, exhibited extensive deterioration. The modification of this Screen Wash pump support slab involved removal of the damaged concrete and installation of steel plates and beams under the slab. Existing conditions and operability determinations for the slab supporting the Screen Wash pumps were documented on inspection reports and Change Request Notices (CRNs.) Bay wall inspections of the CWP 3B2 bay involved visual inspection of concrete surfaces, chipping the concrete at five locations to establish contact with the embedded rebar, and Linear Polarization Resistance Testing under a separate specification. Implementation required that two (3P14 and P14) of the three available Screen Wash pumps be taken out of service during the support slab repair. Repairs were performed only in the 3B2 bay, so that the structural integrity of the adjacent bays were not compromised. Therefore, the repair work did not affect the operability of the Service Water, Component Cooling Water, and Intake Cooling Water pumps located in adjacent bays.

### Safety Evaluation:

Repair work on the Intake Structure was implemented only during Mode 5, Mode 6, or with the reactor defueled. Repairs were implemented so that the structural integrity of the adjacent bays and operability of the Intake Cooling Water pumps supported by these bays were not compromised. The modification did not involve other safety systems or fuel handling operations. Upon completion of the modifications, the structural integrity of the slab was restored to withstand all applicable loads in accordance with the requirements for Class I structures, including pump operating loads and associated components, thereby, meeting the original slab design intent. 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 95-012

UNIT : 3
TURN OVER DATE : 10/01/95

## CONTAINMENT ISOLATION BARRIER TEST ENHANCEMENTS


### Summary:

This Engineering Package modified the containment Penetration Number 10 isolation barrier, located on the Reactor Coolant Drain Tank (RCDT) nitrogen supply flow path line, to allow enhanced local leak rate test capabilities. Specifically, the modification consisted of adding a second outboard manual containment isolation valve that will perform the function of valve PCV-3-1014 as the containment isolation barrier. PCV-3-1014 is a pressure regulating valve, which was also used as a containment isolation. This valve required frequent maintenance work to support successful local leak rate testing on this containment penetration. Also a vent valve was added between this new manual and the existing check valve 3-4657. These additional valves were required to facilitate seat leakage testing of the new manual containment isolation valve and would improve the testability of Penetration 10. Additionally unused test connection valves 3-2053, 3-2060, and 3-2064 were removed on Penetrations 62A, 62B, and 62C, respectively, and the lines were permanently capped. This eliminated a possible containment leakage path to atmosphere. These modifications maintained the current design basis requirements for each of the four penetrations that were modified under this Engineering Package.


### Safety Evaluation:

The modifications performed on the four containment penetrations addressed by this Engineering Package maintained their existing containment penetration design bases and did not introduce any new failure modes not previously analyzed. These modifications ensure that containment integrity will continue to be maintained in compliance with existing containment design bases. 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 these modifications.

## PLANT CHANGE/MODIFICATION 95-019

UNITS : 3 & 4
TURN OVER DATE : 02/13/96

## VEHICLE BARRIER SYSTEM (VBS)

**Summary:**

This Engineering Package established the necessary engineering design documentation for the construction of a "vehicle barrier system" (VBS), which was intended to address the amended requirements of 10 CFR 73, relating to the use of land vehicles to transport personnel and weapons to vital areas of the plant or to transport or locate a land vehicle bomb within destructive proximity of plant vital equipment or structures. These modifications supplement and modify the existing plant security system and include design features to preclude a "land threat vehicle" from penetrating the plant protected area boundary. The modifications consisted of new cable barriers, gates, bollards, and power, control and monitoring equipment/circuitry for the gates located along the plant protected area fence as well as various areas within the adjacent fossil units. The vehicle barrier system was designed to be completely compatible with the existing plant security system fencing, intrusion detection and power supply systems that interfaced with this modification. The modifications were evaluated for seismic interactions with existing safety related structures, systems, and components and were found acceptable with no adverse seismic interactions considered credible.

**Safety Evaluation:**

The design modifications to the plant security system installed under this Engineering Package were classified as not nuclear safety related, except for the those restraints installed on the Unit 3 Switchgear rooms, since these modifications do not perform any safety related functions. However, interactions with safety related structures and systems were evaluated to ensure that no adverse effects would result from these modifications. Consequently, these modifications did not involve an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval under the criteria of 10 CFR 50.59 was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 95-021

UNIT : 3
TURN OVER DATE : 09/15/95

### REPLACEMENT SUBPANEL FOR 3D01 BREAKERS 1 THROUGH 20


**Summary:**

This Engineering Package replaced an electrical distribution subpanel in the DC Load Center 3D01 that contained molded case breakers which were no longer manufactured. The new replacement subpanel contained breakers that are readily available and were expected to remain in production for many years. These new breakers are similar in function and form, but could not have been installed in the original breaker subpanel without modifications. Since the load center was modular in construction and the subpanel was connected by cable to the main load center bus, it was more efficient to replace the entire subpanel instead of replacing individual breakers on the existing subpanel chassis. The new model breakers were a standard commercial model, which were tested and fully qualified for Class 1E nuclear safety related service. This design package also provided spare circuit breakers for maintenance use as spare parts. The replacement activities were scheduled during the last Unit 3 refueling outage after the fuel had been removed from the reactor. Breaker replacements were also performed during a Unit 3 maintenance outage on the 3A 4160 volt AC bus. Temporary System Alterations were implemented for those circuits that required an alternate source of power during the breaker replacements. The evaluation contained in the design package determined that no restrictions were required on the operation of the opposite Unit 4 during breaker replacement implementation. The original style of breakers that were removed but still functioned properly, were recovered for use as spare parts for maintenance of those original style breakers that remained installed in the plant.


**Safety Evaluation:**

The new breakers performed the same functions and meet or exceed the same design and licensing criteria as the original breakers that were replaced. Other evaluations were performed and documented which demonstrated that implementation could be safely performed using temporary power sources. 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 95-023

UNITS : 3 & 4
TURN OVER DATE : 03/10/96

### ABANDONMENT OF CONTINUOUS TUBE CLEANING SYSTEM (COMPONENT COOLING WATER HEAT EXCHANGERS)

**Summary:**

This Engineering Package provided the design change documentation required for abandonment of the Continuous Tube Cleaning System (CTCS) originally installed on the Component Cooling Water (CCW) system heat exchangers. The tube cleaning system was ineffective during the startup testing period following installation and did not maintain an adequate level of tube cleanliness by reducing or preventing heat exchanger fouling and the buildup of calcium carbonate deposits on the heat exchanger tubes. The failure of the CTCS was attributed to three principal factors: (1) the inadequate wiping action of the non-abrasive balls as a result of an inadequate pressure drop across the heat exchangers; (2) the accelerated erosion of the copper alloy tubes prevented frequent operation of the system; and (3) the poor condition of the tube walls. Due to concerns associated with tube erosion and the effect of tube cleaning on ICW flows while in a single pump accident configuration, this tube cleaning system was rarely placed into service. Tube cleanliness of the Component Cooling Water heat exchangers is required to ensure that design basis accident heat loads can be removed by design flows from the Intake Cooling Water (ICW) system. If this tube cleaning system had achieved an appropriate level of cleanliness, this would have reduced the need for periodic manual tube cleaning. Presently, the CCW heat exchangers are being manually cleaned at intervals based on the heat exchanger fouling rate associated with intake canal water temperatures. This cleaning process is currently controlled by several plant procedures.

**Safety Evaluation:**

The tube cleaning system did not serve any safety functions and was not required to support safe shutdown of the plant. The in-place abandonment of this tube cleaning system 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 95-026

UNIT : 3
TURN OVER DATE : 09/21/95

## INSTALLATION OF CANOPY SEAL CLAMP ASSEMBLIES ON SPARE CRDMs

**Summary:**

This Engineering Package provided for the installation of Canopy Seal Clamp Assemblies (CSCAs) to create a new leak-tight boundary on the spare Unit 3 Control Rod Drive Mechanism (CRDM) housings. Units 3 and 4 have experienced several leaks of CRDM housings in the canopy seal weld of the spare Control Rod Drive Mechanism housings. These leaks have resulted in the affected plant being taken off line to implement repairs. To preclude down-time associated with any future leaks on the Unit 3 spare penetrations, the Canopy Seal Clamp Assemblies developed by ABB Combustion Engineering were installed to create a new leak-tight boundary. The CSCA is designed to prevent separation of the clamp and loss of preload on the clamp seal under all service loadings. The CSCA does not impose any unacceptable stresses on the canopy seal of the affected head penetration or in any way jeopardize the integrity of the existing pressure boundary. The CSCA will have adequate clearances when installed and will not adversely interact with other CRDM housings. Hence, no safety related equipment can be adversely impacted or affected by the installation. The CSCA is designed to mechanically encapsulate the canopy seal area of the reactor vessel head nozzle penetration. Therefore, the CSCA also functions as a backup to the non-structural canopy seal weld.

**Safety Evaluation:**

The spare CRDM itself serves no active safety function. The CSCA did not serve as part of the RCS pressure boundary, even though it was mechanically attached to the CRDM housing. The seismic design of the CSCAs installed was reviewed and found acceptable. The materials utilized in the clamp had no adverse interaction with surrounding structures, systems or components. All materials were fully qualified for normal and post-accident environments, and these clamps did not contain any hydrogen generating material and did not impact hydrogen concentration control 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 95-030

UNIT : 3
TURN OVER DATE : 10/04/95

### INSTALLATION AND REMOVAL OF EX-VESSEL NEUTRON DOSIMETRY


**Summary:**

This Engineering Package installed ex-vessel neutron dosimeters in the reactor vessel cavity to measure the neutron flux exposure received by the reactor vessel. The measured flux will be used to calibrate the analytical model used in predicting the reactor vessel fluence. The fluence estimates are used to monitor the reactor vessel embrittlement mechanism and for resolution of the reactor vessel pressurized thermal shock issue. As recommended in the NRC draft Regulatory Guide DG-1025, reactor vessel calculated neutron fluence should be bench-marked with ex-vessel dosimetry. The Turkey Point ex-vessel neutron dosimetry program includes nine stainless steel bead chains with aluminum capsules containing the neutron dosimeters. Each stainless steel bead chain was hung from a bracket that was bolted to the existing bolts on the refueling cavity seal ring floor. The nine bead chains were strategically hung around the periphery of the reactor vessel and each chain contained up to three dosimeter capsules. These capsules were located on each bead chain at three strategic locations: (1) vessel critical weld; (2) core midplane; and (3) axial end of the hafnium rods. Each capsule contains 5 threshold neutron dosimeters selected for a neutron spectrum between 1.0 an 17.0 Mev. The readout of irradiated neutron dosimeters provided the plant measurement data for use in validation of the calculated vessel fluence data, which, in turn, quantifies the effectiveness of the on-going vessel fluence reduction program.


**Safety Evaluation:**

Installation of the chain beads and associated dosimeters was temporary which did not connect to any containment equipment or components vital to plant safety. The chain beads and capsules are environmentally qualified for the containment environment and are made of non-combustible materials. They will remain in the reactor vessel cavity for only one reload cycle. The design package demonstrates that failure of the bead chains or capsules will neither create a different type of accident or reduce any margin of safety. 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 95-032

UNIT : 3
TURN OVER DATE : 09/24/95

## EMERGENCY BUS LOAD SEQUENCER MODIFICATIONS

Summary:

This Engineering Package documented hardware and software modifications which were required for both Unit 3 sequencers in order to eliminate the identified root cause for a failure of Sequencer 3A on November 3, 1994. In addition, this design package also implemented the recommendations from the Independent Assessment Team (IAT) report. Software changes were developed to address the test logic inhibit defect and the containment spray pump start logic defect. Minor hardware and software changes were also implemented as a result of the elimination of the sequencer auto test feature to address sequencer failure modes previously detected by auto test. The emergency bus load sequencers perform two major functions; these are "bus stripping" and "load sequencing." The bus stripping feature trips all the breakers, strips the buses of loads powered from the 4160 volt AC buses, and provides a start signal to the Emergency Diesel Generators (EDGs). The load sequencing feature sequentially closes safety equipment breakers to energize selected equipment in response to postulated design basis accidents. For implementation of these changes, only one sequencer at a time was modified, and these changes were performed while Unit 3 was in either operational Mode 5, 6, or defueled.

Safety Evaluation:

The modifications implemented under this Engineering Package did not affect the sequencers' design basis function to energize engineered safety features in response to design basis accidents. Similarly, there were no changes to the sequencer load block initiation, to response modes, or to load block timing. This Engineering Package evaluated all the changes implemented for the emergency bus load sequencers and demonstrated that each change would not have any adverse effect on plant safety or plant operations, and would not require any changes in plant Technical Specifications. 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 95-033

UNIT : 4
TURN OVER DATE : 03/25/96

## EMERGENCY BUS LOAD SEQUENCER MODIFICATIONS

### Summary:

This Engineering Package documented hardware and software modifications which were required for both Unit 4 sequencers in order to eliminate the identified root cause for a failure of Sequencer 3A on November 3, 1994. In addition, this design package also implemented the recommendations from the Independent Assessment Team (IAT) report. Software changes were developed to address the test logic inhibit defect and the containment spray pump start logic defect. Minor hardware and software changes were also implemented as a result of the elimination of the sequencer auto test feature to address sequencer failure modes previously detected by auto test. The emergency bus load sequencers perform two major functions; these are "bus stripping" and "load sequencing." The bus stripping feature trips all the breakers, strips the buses of loads powered from the 4160 volt AC buses, and provides a start signal to the Emergency Diesel Generators (EDGs). The load sequencing feature sequentially closes safety equipment breakers to energize selected equipment in response to postulated design basis accidents. For implementation of these changes, only one sequencer at a time was modified, and these changes were performed while Unit 4 was in either operational Mode 5, 6, or defueled.

### Safety Evaluation:

The modifications implemented under this Engineering Package did not affect the sequencers' design basis function to energize engineered safety features in response to design basis accidents. Similarly, there were no changes to the sequencer load block initiation, to response modes, or to load block timing. This Engineering Package evaluated all the changes implemented for the emergency bus load sequencers and demonstrated that each change would not have any adverse effect on plant safety or plant operations, and would not require any changes in plant Technical Specifications. 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 95-034

UNITS : 3 & 4
TURN OVER DATE : 08/14/95

### MODIFICATION OF N-35/N-36 REACTOR TRIP BISTABLE RESET VALUE

**Summary:**

This Engineering Package modified the value at which the Nuclear Instrumentation System (part of the Reactor Protection System-RPS) intermediate range (IR) N-35 and N-36 reactor trip bistable modules reset during power descensions. The value at which the bistables (NC-206) reset was 12.5% power, as determined by the IR channels. During power ascensions, this value was determined to be unnecessarily close to the value at which the P-10 deactivation allows the IR trip to occur, i.e., at 8% power, as determined by the Power Range (PR) channels. The small margin allowance that existed between the IR bistable reset and the P-10 deactivation could have potentially caused an unnecessary reactor trip due to a late IR bistable reset, an early P-10 deactivation, or an IR/PR power mismatch. This design package, therefore, modified the IR bistable reset value to 17.5%. This value provides sufficient margin between the IR bistable reset and the P-10 deactivation to preclude unnecessary reactor trips. The IR channels, including the bistables, are part of the Nuclear Instrumentation System (NIS). While no credit is taken for operation of the reactor trips associated with the IR channels, the IR trip function is required by plant Technical Specifications in order to enhance the overall reliability of the Reactor Protection System.

**Safety Evaluation:**

This design change affected only the point at which the IR bistables reset after having been tripped. This modification did not affect the 25% power trip setpoint of the IR reactor trip bistables. The change in the bistable reset point from 12.5% to 17.5% power did not affect the ability of the IR channels to perform the design basis functions required for tripping the reactor during a reactivity excursion with increasing reactor power to 25%, which originated below the P-10 setpoint. Consequently, this design modification did not adversely affect the design bases for the RPS or plant operational requirements, including Technical Specifications. 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 95-043

UNITS : 3 & 4
TURN OVER DATE : 05/11/95

### REMOVAL OF RCA ACCESS POINT GUARDHOUSE ("SHACK")

**Summary:**

This Engineering Package addressed the necessary design documentation, drawings, specifications, and instructions for the removal of an abandoned 10-foot by 12-foot prefabricated modular structure (previously used as a Radiation Controlled Area (RCA) access point Guardhouse / Shack) and its associated materials, equipment, and appurtenances from the Turbine Building at the 18 ft. elevation. This modular structure at the RCA controlled entry point was initially constructed under PC/M 82-204, to provide a new "Security Controlled RCA Access Point." Subsequent to this function, the facility was used for various functions, the most recent of which was a temporary Cold Chemistry Lab. With the recent completion of the new secondary side sample system upgrades, including a new Cold Chemistry Laboratory, no further use for the abandoned modular structure could be found. As a result, it was decided that the former RCA access point structure and its materials, equipment and appurtenances, should be removed rather than abandoned in place. After the removal of this structure, a new section of RCA boundary fencing was installed to preclude uncontrolled access to the RCA. In addition, this package required the removal of a short section of the fire protection system piping, and the re-installation of a support that was seismically designed and evaluated to preclude seismic II over I interactions.

**Safety Evaluation:**

The modular RCA structure performed no safety functions, and its removal had no affect on the operation, function, or design bases of any safety related structure, system, or component. This Engineering Package evaluated various plant interfaces associated with the removal of the RCA structure, including plant security, fire protection, emergency lighting, inplant communications systems, power supply interfaces, RCA boundary integrity, electrical grounding, and seismic requirements. No interfaces adverse to plant safety or plant operation were identified. These modifications, therefore, did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 95-044

UNIT : 3
TURN OVER DATE : 10/26/95

### UNIT 3 CONTAINMENT EQUIPMENT HATCH ACCESS RAMP EXPANSION


#### Summary:

This Engineering Package addressed the design change documentation required to allow an expansion of the Unit 3 containment equipment hatch access ramp. This modification involved the erection of new retaining wall structures to contain the backfill materials for the expanded ramp area. Other work included the installation of a new section of fencing for the radiation controlled area (RCA), relocation of piping used for Integrated Leak Rate Testing (ILRT), and reconnection of the cathodic protection system. During refueling outages, the Unit 3 containment building access ramp is used to stage equipment and temporary facilities for Health Physics, Security, and plant Maintenance. This ramp was very limited in its ability to stage all the temporary facilities which were required to support a refueling outage. This expansion will allow the installation of additional temporary facilities and assist to expedite the progress of the outage. The expanded sections of ramp were designed to carry all required dead load and vehicular highway H-20 loads. The retaining walls were extended above the finished grade to serve as vehicle guard rails and handrails for personnel safety were also provided.


#### Safety Evaluation:

The additional loads imposed on existing plant structures, and on underground duct banks and piping due to site preparation work associated with this modification package were evaluated and found to be acceptable. Hand excavation was employed around existing under-ground utilities to preclude adverse effects to buried piping and electrical systems. The design for the expanded ramp was adequate for its intended purpose and 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 95-054

UNIT : 3
TURN OVER DATE : 10/27/95

## TEMPORARY CONTAINMENT COOLING DURING REFUELING OUTAGES

### Summary:

This Engineering Package installed a temporary containment cooling system through use of a closed loop chilled water air-conditioning system to reduce containment temperatures and humidity during Unit 3 refueling outages. This system utilizes chilled water units to supply cold water to the coils of the nonsafety related Normal Containment Coolers (NCC). The NCC units transfer the heat from within containment to the chilled water loop, using a portion of the Component Cooling Water (CCW) system piping, and then to two self-contained chilled water units located on the auxiliary building roof. Temporary hoses are used to transport the chilled water to the inlet/outlet of the CCW piping flow path. Two permanently anchored roof penetrations and flanged spool pieces were used to complete the connection to the existing CCW piping. This package also removed the piping spool pieces to the Unit 4 boric acid evaporator and documented the permanent installation of the spool pieces from the existing CCW flange connections to the roof penetrations. During use of this chilled water cooling loop, the CCW supply and return headers to the NCCs were designed to be isolated from the main CCW headers. Consequently, all non-essential equipment on this loop would be isolated while the chilled water loop was in operation. The remainder of the CCW system would be unaffected and remain operable. Temporary cables were used to power the chilled water units from the non-safety related Motor Control Centers 3H and 4H. All systems, equipment and piping were designed to be restored to their normal operating configurations following the refueling outage.

### Safety Evaluation:

These design modifications made no permanent changes affecting the operation of the CCW system, the NCCs or their electrical power supplies. The permanent modifications included the permanent piping from the existing CCW flange to the roof penetration; all other modifications and system realignments for this design package were temporary for use during each refueling outage. Operation of the chilled water system will make use of certain portions of the CCW system, but the remainder of the system will function normally and remain operable. 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 95-058

| | | |
|---|---|---|
| UNITS | : | 3 & 4 |
| TURN OVER DATE | : | 03/19/96 |

## RCA MAINTENANCE WORK AREA


**Summary:**

This Engineering Package provided the necessary design documentation, drawings, specifications, and instructions for conversion of the abandoned radwaste compactor room, that was initially built in the late 1970s, into a new Radiation Control Area (RCA) maintenance work area. This maintenance work area was designated for use as an RCA machine shop, that was constructed as a clean, air-conditioned work space and would be equipped with tools such as a lathe, drill press, grinders, and other tools. The design package also addressed the supporting utilities that were required for this work space. This modification included the removal of the existing abandoned power distribution system and installation of a new distribution system and receptacles that would power the new machine shop tools. Lighting was replaced and new raceways were installed. A new air-conditioning unit was installed along with a two-ton capacity trolley and hoist system. The existing floor drain pit and discharge piping were closed off and the drain pit was filled with grout to prevent any contaminated materials and oil spills from reaching the radwaste building sump system.


**Safety Evaluation:**

A walkdown inspection and review of the previously abandoned radwaste compactor room facility confirmed that no existing safety related functions or installations were associated with this area, and there was no potential for interactions with any safety related structures, systems, or components. Similarly, the conversion of this same room to a new RCA machine shop did not adversely impact any safety functions or interface with any safety related structures, systems or components. None of the new materials and equipment installed under this design package perform any safety related functions, however, these modifications did require the installation of new conduit/supports that were seismically designed/evaluated to preclude seismic category II over I interactions. 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 95-060

| | | |
|---|---|---|
| UNITS | : | 3 & 4 |
| TURN OVER DATE | : | 03/30/96 |

## ELIMINATION OF CRANKING DIESEL TIE AND UNITS 3 & 4 C-BUS INTERLOCKS

### Summary:

This Engineering Package removed the electrical tie between the Cranking Diesel Generators (CDG) or "Black Start Diesels" and the 4160 volt nonsafety C-bus. This modification deleted the Cranking Diesel Generators (CDG) as a backup power source for both units C-bus switchgear. The power cable between C-bus switchgear 3C & 4C was left as a cross-tie, with the cross-tie breakers normally racked out. The Cranking Diesel Generator instrumentation and indication for Units 3 & 4 were removed from control panels in the control room. In addition, the switchgear 3C and 4C breaker interlocks make it possible to power any safety related A or B bus using a C-bus transformer. In addition, the switchgear 3C/4C cross-tie could be used as a backup for the Station Blackout (SBO) tie. These possible configurations were not intended for use during normal plant operation and no credit was taken for their use during analyzed design basis accidents or Station Blackout conditions. The subject line-ups serve to provide additional controlled options for operating personnel should plant conditions warrant their use. The existing inter-tie between the nonsafety related C-Bus and the safety related A or B-bus will remain administratively controlled with all tie breakers normally racked out and locked with control fuses pulled. Similar controls were retained for switchgear 3C/4C cross-tie breakers. The C-Bus loads that presently take credit for the Cranking Diesel Generators as a backup power source were re-powered from other safety related or nonsafety grade power sources.

### Safety Evaluation:

This modification did not alter any existing safety related systems or components and did not add any additional safety related equipment. This Engineering Package addressed electrical system functions, design bases, design and installation criteria (including, control room human factors, seismic qualifications and interactions, electrical separation criteria, and failure modes and effects), and plant restrictions, and concluded that these modifications would not have any adverse effect on plant safety or operation. Consequently, these modifications did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

# PLANT CHANGE/MODIFICATION 95-075

UNIT : 3
TURN OVER DATE : 11/14/95

## REPLACEMENT OF MAIN STEAM SAFETY RELIEF VALVE DISCHARGE PIPING


**Summary:**

This Engineering Package replaced the existing ten-inch diameter discharge piping for each of the twelve Main Steam Safety Valves (MSSV) on Unit 3 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 MSSV's. In addition, permanent drains were provided for the discharge piping. The MSSV's serve to protect the secondary plant system from overpressurizing. Four MSSV's are located on each of the three main steam lines, upstream of the Main Steam Isolation Valves (MSIV's). Based on analyses performed for the plant uprated condition, it was determined that the back pressure on the MSSV's resulting from the mass flow rate at uprate conditions coupled with the unusually long discharge lines would be excessive, which 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 95-084

UNITS : 3 & 4
TURN OVER DATE : 08/18/95

## REACTOR COOLANT PUMP (SPARE) MOTOR REFURBISHMENT/UPGRADE

### Summary:

This engineering package was developed to upgrade a spare motor (Serial Number 1S-86P763) for installation on any of the Unit 3 or Unit 4 reactor coolant pumps as part of a program to improve the reliability and performance of all reactor coolant pumps at Turkey Point. This spare motor was refurbished at the Westinghouse Electro-Mechanical Division facility. This standard factory refurbishment consisted of inspection and maintenance activities performed to the existing design specifications. In addition, modifications were conducted, concurrent with the refurbishment, to ensure consistency with the latest reactor coolant pump technology and to realize additional reliability and availability. Upon completion of the modifications, the motor was assembled, balanced and tested and shipped back to Turkey Point for use as a spare. The motor modifications documented by this design package included the following: (1) spare resistance temperature detector modifications; (2) flywheel pawl modifications; and (3) trunnion adapter plate modifications. These modifications did not change the design function of any components, and did not affect reactor coolant pump coastdown characteristics.

### Safety Evaluation:

Under this design package, the spare motor was refurbished and upgraded by the original vendor under 10 CFR 50, Appendix B Quality Assurance Program. No modifications were made to any pump or mounting components. The original design for seismic interactions and missile generation remained unaffected. None of the modifications made under this Engineering Package affected the ability of the motor flywheel from functioning for flow coastdown nor did they increase the probability of occurrence or consequences of a reactor coolant pump shaft seizure or loss of flow accident. 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 95-087

UNIT : 4
TURN OVER DATE : 04/05/96

## ROD CONTROL CRDM TIMING CHANGES


### Summary:

This Engineering Package was developed to implement Control Rod Drive Mechanism (CRDM) current order timing changes and additional rod control system surveillances as recommended by Westinghouse in their Technical Bulletin NSD-TB-94-05-R0, which was issued to address the Salem Plant Rod Control System event on May 27, 1993. During this event at the Salem Plant, the Rod Cluster Control Assemblies (RCCA) responded inconsistently to corrupted current orders, such that one control rod moved out while inward motion was demanded by the control system. Responding to this event, the NRC issued Generic letter (GL) 93-04 requesting a response to the events at Salem. FPL responded to GL 93-04 by performing a nuclear fuels analysis that demonstrated that existing DNB margins in the Unit 3 and 4 fuel designs were adequate to prevent exceeding fuel design limits on an asymmetrical rod withdrawal event, such as had occurred at Salem. The modifications addressed in this Engineering Package consisted of changing diode connections on each slave cycler decoder printed circuit board located in the Rod Control system logic cabinet. Additional surveillances of the Control Rod System were also implemented to detect single failures which do not affect rod motion, and therefore, are not detectable by routing rod insertion/withdrawal.


### Safety Evaluation:

The Rod Control system is not required to function to achieve or maintain safe shutdown or mitigate the consequences of design basis accidents. However, this system is important to safety, since it controls reactor power during normal operation, and its failure can initiate design basis accidents documented in the Updated FSAR. This modification ensured that a Salem type failure will cause the affected rod control group/bank to insert rather than withdraw, which has been shown to be bounded by the consequences of other design basis accident events already analyzed in the Updated FSAR. Therefore, with the revised CRDM timing modifications, the current licensing basis analyses remain bounding and 1967 Proposed GDC 31 continues to be met for all single failures within the control rod system. Consequently, these CRDM 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 this modification.

## PLANT CHANGE/MODIFICATION 95-096

UNIT : 3
TURN OVER DATE : 12/22/95

## UNIT 3 EDG CONTROL WIRING MODIFICATIONS


**Summary:**

This Engineering Package was developed to eliminate an identified "relay race" between relays SF-X, TD2 and SF in the Unit 3 Emergency Diesel Generators'(EDG) control circuitry, which could prevent lockout of an Emergency Diesel Generator if it had failed to start. These design changes also modified EDG circuitry, such that, the main bearing (lube) oil low pressure trip and alarm circuits would be enabled when an EDG "Normal Stop" is initiated. Previously, there was no lube oil low pressure trip or alarm capability during a normal EDG shutdown sequence. As a result of these modifications, the engine run meter will record the 20 minute idle EDG run time, which was previously not recorded during a normal shutdown sequence. This design deficiency applied to non-emergency EDG operation only; lube oil low pressure trip and alarm protection functions were desirable for equipment protection during non-emergency run conditions. The modifications in this design package involved relatively minor internal wiring changes in the EDG control panels. No new components or cables were required. Implementation of these modifications was allowed in any plant operating mode, but concurrent outages for EDGs 3A and 3B were prohibited.


**Safety Evaluation:**

This modification did not add any new devices, and no changes were made to the emergency operation of the EDGs. The wiring changes served to restore the original nonsafety functions to the EDG control circuitry. 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 diesel control logic, 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 the design package evaluation, these modifications did not have any adverse effects on plant safety or plant 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 these modifications.

## PLANT CHANGE/MODIFICATION 95-100

UNIT : 4
TURN OVER DATE : 07/01/96

## RTDP RELATED RPS/ESFAS SETPOINT CHANGES


### Summary:

This Engineering Package documents implementation of the Westinghouse Revised Thermal Design Procedure (RTDP) Instrument Uncertainty Methodology (WCAP-13719) to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) for Unit 4. A Plant Licensing Amendment (PLA) was approved by the NRC to allow the RTDP to be incorporated for use in calculating the Departure Form Nucleate Boiling Ratio (DNBR) limits for Units 3 and 4. The RTDP method-ology statistically combines the uncertainties of pressurizer pressure, reactor coolant system (RCS) temperature, reactor power and RCS flow with the DNBR correlation uncertainty to calculate new DNBR limits. By utilizing the RTDP methodology, increased margin was obtained between the DNBR design and safety analysis limits, which can be utilized to offset thermal uprate and unanticipated DNBR penalties. As a result, the over-temperature (OT$\Delta$T) and overpower (OP$\Delta$T) $\Delta$T trip setpoint, and associated uncertainties were changed. $\Delta$T, $T_{HOT}$, and $T_{COLD}$ instrumentation ranges were also affected. This Engineering Package also changed the setpoint and uncertainty parameters for Steam Generator (S/G) Low and Low-Low Level trip functions as a result of the re-evaluation of the process measurement parameter. These changes provided a more accurate assessment of the PMA term and allow for additional operating margin to reduce the possibility of spurious reactor trips during low power transients. The setpoint related changes were not a direct result of RTDP, but were included in the accident analysis evaluations to support application of the RTDP methodology. Another change involving a time delay for the reactor coolant pump undervoltage trip was also addressed by this package.


### Safety Evaluation:

These RPS/ESFAS setpoint modifications did not physically alter any equipment important to safety. This modification did not change the operation, function or design bases of any structure, system or component important to safety as described in the SAR and no new hazards were created that could cause an accident different from those previously analyzed. Consequently, these RPS/ESFAS setpoint changes did not involve an unreviewed safety question. Due to the changes in RPS/ESFAS setpoints, NRC approval was obtained for the Technical Specification changes in advance of Engineering Package implementation. Therefore, specific NRC approval was not required for implementation of the Engineering Package modifications.

## PLANT CHANGE/MODIFICATION 95-101

UNIT : 4
TURN OVER DATE : 03/17/96

## INSPECTION, REPAIR AND MODIFICATION OF THE UNIT 4 INTAKE STRUCTURE


### Summary:

This engineering package was prepared to restore the Unit 4 Intake Structure slab, supporting Screen Wash Pump 4P14, to its original or equivalent design configuration, thereby ensuring adequate long-term performance of the structure. Concrete walls for the Circulating Water Pump (CWP) 4A1 bay were inspected for any active corrosion of the steel reinforcement within the walls. The concrete slab for the CWP 4A1 bay, which supports the Screen Wash pump 4P14, exhibited extensive deterioration. The modification of this Screen Wash pump support slab involved removal of the damaged concrete and installation of steel plates and beams under the slab. Existing conditions and operability determinations for the slab supporting the Screen Wash pumps were documented on inspection reports and CRNs. Bay wall inspections of the CWP 4A1 bay involved visual inspection of concrete surfaces, chipping the concrete at four locations to establish contact with the embedded rebar, and Linear Polarization Resistance Testing under a separate specification. Implementation required that one (4P14) of the three available Screen Wash pumps be taken out of service during the support slab repair. Repairs were performed only in the 4A1 bay, so that the structural integrity of the adjacent bays was not compromised. Therefore, the repair work did not affect the operability of the Service Water pumps, Component Cooling Water pumps, and Intake Cooling Water pumps located in adjacent bays.


### Safety Evaluation:

Repair work on the Intake Structure was implemented with two screen wash pumps available at all times. Repairs were implemented so that the structural integrity of the adjacent bays and operability of the Intake Cooling Water pumps supported by these bays were not compromised. The modification did not involve other safety systems or fuel handling operations. Upon completion of the modifications, the structural integrity of the slab was restored to withstand all applicable loads in accordance with the requirements for Class I structures, including pump operating loads and associated components, thereby, meeting the original slab design intent. 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 95-105

UNIT : 4
TURN OVER DATE : 03/23/96

## INSTALLATION OF CANOPY SEAL CLAMP ASSEMBLIES ON SPARE CRDMs


**Summary:**

This Engineering Package provided for the installation of Canopy Seal Clamp Assemblies (CSCAs) to create a new leak-tight boundary on the spare Unit 4 Control Rod Drive Mechanism (CRDM) housings. Units 3 and 4 have experienced several leaks of CRDM housings in the canopy seal weld of the spare Control Rod Drive Mechanism housings. These leaks have resulted in the affected plant being taken off line to implement repairs. To preclude down-time associated with any future leaks on the Unit 4 spare penetrations, the Canopy Seal Clamp Assemblies developed by ABB Combustion Engineering were installed to create a new leak-tight boundary. The CSCA is designed to prevent separation of the clamp and loss of preload on the clamp seal under all service loadings. The CSCA does not impose any unacceptable stresses on the canopy seal of the affected head penetration or in any way jeopardize the integrity of the existing pressure boundary. The CSCA will have adequate clearances when installed and will not adversely interact with other CRDM housings. Hence, no safety related equipment can be adversely impacted or affected by the installation. The CSCA is designed to mechanically encapsulate the canopy seal area of the reactor vessel head nozzle penetration. Therefore, the CSCA also functions as a backup to the non-structural canopy seal weld.


**Safety Evaluation:**

The spare CRDM itself serves no active safety function. The CSCA did not serve as part of the RCS pressure boundary, even though it was mechanically attached to the CRDM housing. The seismic design of the CSCAs installed was reviewed and found acceptable. The materials utilized in the clamp had no adverse interaction with surrounding structures, systems or components. All materials were fully qualified for normal and post-accident environments, and these clamps did not contain any hydrogen generating material and did not impact hydrogen concentration control 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 95-130

UNIT : 3
TURN OVER DATE : 09/30/95

### MODIFICATION OF LOW PRESSURE AUTO START FOR CCW PUMPS


**Summary:**

This Engineering Package modified the Component Cooling Water (CCW) pump control circuitry associated with an automatic pump start on "low header pressure" to eliminate credible mechanistic circuit failures, which could have resulted in an undesirable CCW pump start. While performing CCW flow balancing analyses to meet new requirements imposed by Thermal Uprate Project criteria, certain existing CCW flow configurations were identified with maximum calculated CCW flows which could have exceeded flow limitations established for certain CCW and safeguards components. To preclude the possibility of these events, this Engineering Package reduced the CCW pump low pressure actuation setpoint and altered automatic start time delays to ensure that undesirable automatic starts during expected system alignments and transients did not occur. These modifications also ensure that any credible mechanistic failure in the CCW header low pressure start circuitry will not cause an automatic start of an additional CCW pump. These modifications involved changing a normally energized relay in the low header pressure start circuit to a normally de-energized state. The setpoint for the low pressure pump start was reduced from 60 psig to 35 psig, and the start time delays for CCW pumps 3A and 3B were increased from 5 and 15 seconds to 10 and 20 seconds, respectively.


**Safety Evaluation:**

The CCW pump low pressure automatic start feature is not a safety related function, and no credit is taken for the low header pressure start signal to achieve or maintain safe shutdown, to prevent or mitigate the consequences of accidents, or protect equipment required for safe shutdown or accident mitigation. Only a single pressure switch and auxiliary relay are used to perform the pump low pressure auto-start function. Therefore, this circuitry is not required to satisfy single failure criteria. The modifications documented in this design package do not have any adverse effect on plant safety or plant operation. 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 95-132

UNIT : 4
TURN OVER DATE : 03/01/96

## MODIFICATION OF LOW PRESSURE AUTO START FOR CCW PUMPS

### Summary:

This Engineering Package modified the Component Cooling Water (CCW) pump control circuitry associated with an automatic pump start on "low header pressure" to eliminate credible mechanistic circuit failures, which could have resulted in an undesirable CCW pump start. While performing CCW flow balancing analyses to meet new requirements imposed by Thermal Uprate Project criteria, certain existing CCW flow configurations were identified with maximum calculated CCW flows which could have exceeded flow limitations established for certain CCW and safeguards components. To preclude the possibility of these events, this Engineering Package reduced the CCW pump low pressure actuation setpoint and altered automatic start time delays to ensure that undesirable automatic starts during expected system alignments and transients did not occur. These modifications also ensure that any credible mechanistic failure in the CCW header low pressure start circuitry will not cause an automatic start of an additional CCW pump. These modifications involved changing a normally energized relay in the low header pressure start circuit to a normally de-energized state. The setpoint for the low pressure pump start was reduced from 60 psig to 35 psig, and the start time delays for CCW pumps 4A and 4B were increased from 5 and 15 seconds to 10 and 20 seconds, respectively.

### Safety Evaluation:

The CCW pump low pressure automatic start feature is not a safety related function, and no credit is taken for the low header pressure start signal to achieve or maintain safe shutdown, to prevent or mitigate the consequences of accidents, or protect equipment required for safe shutdown or accident mitigation. Only a single pressure switch and auxiliary relay are used to perform the pump low pressure auto-start function. Therefore, this circuitry is not required to satisfy single failure criteria. The modifications documented in this design package do not have any adverse effect on plant safety or plant operation. 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 95-133

UNIT : 3
TURN OVER DATE : 10/03/95

### ADD ACCUMULATOR TO ECC CCW OUTLET ISOLATION VALVES


**Summary:**

In order to resolve a potential issue associated with Component Cooling Water (CCW) system flows, this Engineering Package introduced a time delay between loss of instrument air and the operation of fail-open Emergency Containment Cooler (ECC) CCW outlet isolation valves. The subject CCW outlet isolation valves are air-operated valves located in the CCW return header from their respective ECCs. These valves are normally closed, and open for an ECC actuation signal, or upon loss of instrument air, i.e., when pressure drops below 45 psig. While this failure mode satisfies fail-safe criteria, it may also result in the potential for CCW flows to be diverted to equipment which does not require this CCW flow. The CCW diversion flow paths could potentially cause CCW flows to exceed the design flow of the CCW/ICW heat exchangers. Following a loss of instrument air pressure, a reasonable time was established for operator actions to address the potential for these CCW flow conditions, i.e., to ensure that the appropriate number of CCW pumps, heat exchangers, and loads are balanced in service. This PC/M added an isolated air accumulator to the control valve pneumatics to reduce the rate of decay of air pressure at the control valves and ensure that they can remain closed for at least 20 minutes following a loss of instrument air. The design package accomplished this objective through the addition of a check valve and accumulator in the air supply line to each control valve. This did not, however, affect the capability of the control valves to open on demand to allow the operation of the ECCs post-accident.


**Safety Evaluation:**

The accumulators tanks, tubing, and mounting configurations were evaluated and found adequate for all applicable loads, including seismic. The accumulators met applicable ASME requirements, and other components exceeded the nominal design rating of the instrument system. Based on the criteria evaluated in the design package these modifications did not have any adverse impact on safety related structures, systems, or components and did not affect plant operations. 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 95-135

UNIT : 3
TURN OVER DATE : 10/05/95

## AUXILIARY SPRAY CONTROL VALVE CV-3-311 THERMAL RELIEF FUNCTION ENHANCEMENTS

### Summary:

This Engineering Package installed a relief valve and associated piping and fittings to assure that proper thermal relief protection was provided for the Unit 3 Regenerative Heat Exchanger (RHX) and connected components. This installation was required after it was discovered that proper thermal relief protection could not be assured by the component originally intended to perform this function, the Auxiliary Spray Control Valve CV-3-311. During unit operation, it was discovered that CV-3-311 was not set properly to provide thermal over-pressure protection for the RHX. An evaluation was completed and compensatory actions were established to assure the functionality of the RHX until the valve could be tested and reset. While attempting to reset this valve during a refueling outage, it was determined that it could not be reset to perform its intended thermal relief function while also providing isolation during routine operation. Specifically, the valve could not be set to reliably open at a differential pressure of 250 psid and still be closed by its spring force alone. This design package resolved this issue by installing a relief valve (designated RV-3-311) in the charging header downstream and close to the containment isolation check valve. The new relief valve will discharge to a common line that collects valve packing leakoff and discharge from various relief valves, and routes them to the Pressurizer Relief Tank (PRT).

### Safety Evaluation:

This Engineering Package evaluated and documented the new relief valve selection and compatibility with the thermal relief functions that it would be required to perform. The new valve was evaluated for stresses from dead weight, thermal, pressure, seismic, and dynamic effects and was found to meet the Updated FSAR requirements for Class I systems. Potential failure modes were reviewed and plant operating restrictions and precautions were identified. This design change package concluded that the proposed modifications would have no adverse affect on plant safety or plant operations. 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 95-156

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

## AUXILIARY SPRAY CONTROL VALVE CV-4-311 THERMAL RELIEF FUNCTION ENHANCEMENTS

### Summary:

This Engineering Package installed a relief valve and associated piping and fittings to assure that proper thermal relief protection was provided for the Unit 4 Regenerative Heat Exchanger (RHX) and connected components. This installation was required after it was discovered that proper thermal relief protection could not be assured by the Unit 4 Auxiliary Spray Control Valve CV-4-311. As the result of an investigation into the equivalent Auxiliary Spray Control Valve on Unit 3 (CV-3-311), it was discovered that CV-4-311 was not set properly and could not reliably provide thermal over-pressure protection for the Unit 4 RHX. The Unit 4 valve was of an identical design to the equivalent valve on Unit 3. An evaluation was completed and compensatory actions were established to assure the functionality of the Unit 4 RHX until the valve could be tested and reset. While attempting to reset the equivalent Unit 3 valve during a refueling outage, it was determined that it could not be reset to perform its intended thermal relief function while also providing isolation during routine operation. Specifically, the valve could not be set to reliably open at a differential pressure of 250 psid and still be closed by its spring force alone. This design package resolved this issue by installing a relief valve (designated RV-4-311) in the charging header downstream and close to the containment isolation check valve. The new relief valve will discharge to a common line that collects valve packing leakoff and discharge from various relief valves, and routes them to the Pressurizer Relief Tank (PRT).

### Safety Evaluation:

This Engineering Package evaluated and documented the new Unit 4 relief valve selection and compatibility with the intended thermal relief functions. The new valve was evaluated for stresses from dead weight, thermal, pressure, seismic, and dynamic effects and was found to meet the Updated FSAR requirements for Class I systems. Potential failure modes were reviewed and plant operating restrictions and precautions were identified. This design package concluded that theses modifications would have no adverse affect on plant safety or plant operations. Consequently, these modifications did not constitute an unreviewed safety question or require changes to plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of these modifications.

## PLANT CHANGE/MODIFICATION 95-159

UNIT : 4
TURN OVER DATE : 03/31/96

## ENHANCEMENT OF 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 RPS to provide 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 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, 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.

### 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 time 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 95-172

UNIT : 4
TURN OVER DATE : 03/26/96

## UNIT 4 BORON INJECTION TANK BYPASS MODIFICATION


### Summary:

This Engineering Package implemented a new simplified Safety Injection flow path, which accommodated the previous abandonment of the Unit 4 Boron Injection Tank (BIT) and associated piping and equipment. The BIT was an original plant design feature which was abandoned in place based on a re-analysis of the main steam line break (MSLB) performed for replacement of the steam generators in the early 1980s; this MSLB re-analysis removed the need for high-concentration boric acid injection and was implemented as result of a shift in NRC regulatory philosophy during this period. During removal of abandoned heat tracing and insulation from boric acid supply piping in the common boric acid batching room, multiple through-wall leaks in stainless steel piping and components were discovered. These were attributed to stress corrosion cracking and were addressed and documented in plant Condition Reports. Based on an engineering review of other systems, the BIT Safety Injection piping was identified as potentially subject to the same stress corrosion cracking failure mechanisms. Corrective actions included inspection and replacement of this piping. Removing the BIT from the pressurized flow path of the Safety Injection system and re-routing of the subject piping was determined to be 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 4 BIT and associated piping/equipment, by installing a permanent section of qualified piping to bypass the BIT and associated piping.


### Safety Evaluation:

This design package required implementation while the unit remained in operational Modes 5, 6, or defueled. It also established other configuration restrictions for implementation, since the High Head Safety Injection system is shared between the two units. The design package evaluated system design bases and functions, design and installation criteria, and other applicable criteria, and concluded that these system changes would not have any adverse impact on plant safety or plant operation. 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.

## SECTION 2

## SAFETY EVALUATIONS

## SAFETY EVALUATION JPN-PTN-SEEJ-88-042
REVISIONS 4, 5, and 6

| | | | |
|---|---|---|---|
| UNIT | : | | 4 |
| APPROVAL DATES | : | Rev.4 | 02/07/96 |
| | | Rev.5 | 03/05/96 |
| | | Rev.6 | 03/18/96 |

## 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 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.

Revisions 4, 5 and 6 revise this evaluation to update the electrical design configuration requirements and impose new restrictions to preclude emergency diesel overloading and interties.

### 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 FSAR 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 safe question or require changes to plant Technical Specificatio Therefore, prior NRC approval was not required for implementa

## SAFETY EVALUATION JPN-PTN-SEEJ-89-085
REVISION 9

UNIT : 3
APPROVAL DATE : 08/17/95

## DE-ENERGIZATION OF UNIT 3 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 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 EDG from service as the result of a Unit 3 4160 volt bus de-energization. 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 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 9 of this evaluation incorporates additional requirements to allow for the operation of the EDG 3A air compressor during the 3A bus outage.


**Safety Evaluation:**

This safety evaluation addressed the technical and licensing requirements for the de-energization of each Unit 3 4160 volt bus and concluded that the proposed plant configuration and mode of operation was bounded by the Technical Specifications and did not change the analysis of accidents addressed in the FSAR or the results and conclusions of any previous safety evaluations. The actions or precautions identified and evaluated in this safety evaluation did not have any adverse effect on plant safety or plant operations. The actions or 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.

**SAFETY EVALUATION JPN-PTN-SECS-92-020**
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 10/19/95

## 10 CFR 50.59 SAFETY EVALUATION FOR SPECIFICATION SPEC-C-005: COMPONENT MOUNTING AND SUPPORTS


### Summary:

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-C-005 in the maintenance process, in lieu of the current practice which requires that a Plant Change Modification (PC/M) package be issued and implemented for such cases. It also demonstrated that the specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-C-005 was developed to provide generic component mounting instructions and support details. These mounting instructions and support details could be used in conjunction with design output documents presently used in the procurement/maintenance process (e.g., Procurement Technical Evaluations, Item Equivalency Evaluations, or Plant Work Orders) to install replacement components weighing less than 50 pounds. This generic engineering specification was applicable to all plant safety classifications.

Revision 1 of this evaluation addresses the use of a Specification Clarification or Change Sheet provided within the generic specification to obtain Engineering approval for the mounting details that are not pre-approved in the specification.


### Safety Evaluation:

The safety evaluation concluded that the provisions of this generic specification for component mounting and supports met all technical and licensing requirements from the Turkey Point Unit 3 & 4 Updated FSAR and would have no adverse impact on plant operations or existing plant structures. In addition, the evaluation addressed spatial interactions, maintenance accessibility, and materials requirements, including the potential for hydrogen generation within containment. Use of the mounting instructions and support details in accordance with 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 JPN-PTN-SEMS-92-041
REVISION 4 and 5

| | | | |
|---|---|---|---|
| UNIT | : | | 3 |
| APPROVAL DATES | : | Rev.4 | 09/01/95 |
| | | Rev.5 | 09/05/95 |

## SAFETY EVALUATION FOR ICW PIPING INSPECTION AND REPAIR


**Summary:**

Periodically, crawl-through inspections of the Intake Cooling Water (ICW) system piping are conducted to evaluate the condition of the piping. Concurrently, patching of the concrete lining and/or repair of the cast iron piping is performed. This evaluation was developed to address all crawl-through inspection and repair evolutions conducted on the Unit 3 ICW headers. Only cement-lined cast/ductile iron piping with a nominal size greater than 24-inches was included in the scope of the inspections. The purpose of the inspections was to verify the integrity of the pressure boundary from inside the pipe, arrest any significant corrosion that may be occurring, and to validate the integrity of previous repairs. In addition, the location of all patched or repaired areas was mapped and documented. This mapping of patched/repaired areas will allow for the stress analysis of record to be periodically updated to precisely indicate the adequacy of the ICW system piping with respect to Updated FSAR design basis criteria. Further, the rate of corrosion may be established which would allow for an optimum inspection interval to be established. The specific purpose of the safety evaluation was to demonstrate that the special configuration(s), into which the plant was placed for the piping inspections and repair activities, would not result in a condition with an adverse effect on plant safety and operation.

Revision 4 of this evaluation incorporated a distinction between a system's operability and its ability to perform its intended support function. Revision 5 removed certain plant restrictions and permitted crawl-through inspections to be performed in accordance with existing plant procedures and guidance.


**Safety Evaluation:**

The ICW configurations evaluated do not change the design of any safety related equipment nor would they create any additional safety hazards for safety related equipment. Required redundancies are maintained during the inspections so no new active failure modes were introduced by these inspections. The actions or temporary plant measures 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 measures identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SECJ-93-001
REVISION 2

UNITS : 3 & 4
APPROVAL DATE : 06/22/95

# SAFETY EVALUATION FOR SPECIFICATION SPEC-C-013: INSTALLATION GUIDELINES FOR MISCELLANEOUS NON-SYSTEM RELATED ITEMS ON EXISTING STRUCTURES


## Summary:

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-C-013 in the maintenance process, in lieu of the current practice which required that a Plant Change/Modification (PC/M) package be issued and implemented for such cases. It also demonstrated that the Specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-C-013 was developed to provide generic guidance for the installation of storage racks, fire extinguisher, storage cabinets and other miscellaneous items, which are not part of a plant system. These guidelines can be used directly for those installations covered by the support details provided in Appendix B of the specification, or in conjunction with additional guidance provided by Engineering with the form provided in Appendix C.

Revision 1 imposed additional restrictions on the use of this generic evaluation and made various administrative changes. Revision 2 of the evaluation incorporated Revision 2 of the Engineering Specification SPEC-C-013 and added details for a typical ladder.


## Safety Evaluation:

The evaluation indicated that installation guidelines and support details met the Turkey Point Updated FSAR requirements for the applicable installation safety classifications. Existing structures, which provide the foundation for the supports, were not adversely affected. Finally, the containment accident analysis was not adversely affected. The actions or proposed guidelines 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 proposed guidance identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SECJ-93-002
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 04/16/96

## SAFETY EVALUATION FOR SPECIFICATION SPEC-C-014: GUIDELINES FOR INSTALLATION AND USE OF RIGGING ATTACHMENTS


### Summary:

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-C-014 in the maintenance process, in lieu of the current practice which requires that a Plant Change Modification (PC/M) package be issued and implemented for such cases. It also demonstrated that the specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-C-014 was developed to provide generic details for lifting lugs to be used for small loads (e.g., pump motors, valves, pipe spools). The specification also provided guidance for fabrication, installation, and use of the lifting lugs. The applicable lug configuration would be selected from this specification by the engineer or implementer to suit existing field conditions and lifting requirements. The installation of new lifting lugs shall be submitted to Engineering for review and approval on the form provided in Appendix B to the specification. Once initial approval is obtained, the lifting lugs may be used for future lifts without Engineering approval, provided that the technical requirements of the specification are satisfied. Deviations from the specification technical requirements would also be submitted to Engineering for approval.

Revision 1 served to distinguish between permanent and temporary lifting attachments and refined the Engineering approval process for both types of attachments.


### Safety Evaluation:

The evaluation indicated that all lifting lug installations will meet the Turkey Point Unit 3 & 4 Updated FSAR requirements for the applicable installation safety classifications. Existing structures, which will provide support for the lifting lugs, will not be adversely affected. The actions or proposed guidance 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 proposed guidance identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SEFJ-93-013
REVISION 2

UNITS : 3 & 4
APPROVAL DATE : 07/13/95

## SAFETY EVALUATION OF SPENT FUEL POOL (SFP) COUPON SURVEILLANCE PROGRAMS


### Summary:

This safety evaluation was developed to establish a technical basis for an enhanced spent fuel rack Boraflex verification program, using Blackness testing techniques in place of continued reliance on the "Boraflex Coupon Surveillance Program." Blackness testing is a technique used to measure the level of neutron absorption (degree of blackness) of the spent fuel racks with Boraflex or other neutron absorbing material(s) installed. High density spent fuel storage racks were installed in the spent fuel pools (SFP) of Units 3 and 4. The design of the rack includes the use of Boraflex material, which is a strong neutron absorption material used to maintain the spent fuel in the pool in a subcritical array. Boraflex degradation has been an issue since 1987, and extensive industry study of the degradation of the Boraflex material has utilized the techniques associated with blackness testing and analysis of coupons. The technical review of the SFP Coupon Surveillance Program at Turkey Point was performed and completed. Review of the results indicated that coupon surveillances had little merit with respect to inservice performance of Boraflex panels in the racks. Therefore, the ongoing coupon surveillance program for both units can be eliminated. An enhanced Boraflex verification program has been developed to replace the existing in-service coupon surveillance program. Details of this enhanced blackness testing program are described in this safety evaluation.

Revision 2 modified the 50.59 safety evaluation to address cancellation of the coupon surveillance program.


### Safety Evaluation:

This evaluation examined the proposed blackness testing methodology and elimination of the Boraflex coupon surveillances, and it concluded that this would not violate Technical Specification requirements for the spent fuel pool, and would not alter any margin of safety associated with the prevention and mitigation of fuel handling accidents. Consequently, the use of blackness testing methods and elimination of the coupon surveillances 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 enhanced blackness testing methodogy and elimination of the coupon surveillances.

## SAFETY EVALUATION JPN-PTN-SECJ-93-014
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 04/16/96

## 10 CFR 50.59 SAFETY EVALUATION FOR SPECIFICATION SPEC-C-017: SMALL BORE PIPING SUPPORTS


### Summary:

This evaluation provided the basis for the acceptability of using Engineering Specification SPEC-C-017 in the maintenance process, in lieu of the current practice, which requires that a Plant Change Modification (PC/M) package be issued and implemented for such cases. It also demonstrated that the specification met all technical and licensing requirements for the Turkey Point Nuclear Units. Engineering Specification SPEC-C-017 was developed to provide generic installation instructions and details for Safety Related, Quality Related, and Non-Nuclear Safety small bore pipe supports. These installation instructions and support details could be used as specified within Engineering design output documents, such as PC/Ms. In addition, maintenance, repair or replacement activities may be performed on existing small bore pipe supports using the specified standard supports in conjunction with the additional guidance provided by Engineering via the "Maintenance Request Approval" (MRA) process.

Revision 1 allowed the installation of additional supports on Non-Nuclear Safety Related piping. This provision was added for consistency with other generic Engineering specifications, such as, SPEC-C-024,"Large Bore Pipe Supports" and SPEC-C-021,"Tubing and Tubing Supports."


### Safety Evaluation:

The safety evaluation concluded that the provisions of this generic specification for small bore pipe supports met all technical and licensing requirements from the Turkey Point Unit 3 & 4 Updated FSAR and would have no adverse impact on plant operations or existing structures and/or support foundations. In addition the evaluation addressed spatial interactions, maintenance accessibility, and materials requirements, including the potential for hydrogen generation within containment. 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 JPN-PTN-SENS-93-021
REVISIONS 2 and 3

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|---|---|---|---|
| UNIT | : | | 4 |
| APPROVAL DATES | : | Rev.2 | 12/09/94 |
| | | Rev.3 | 06/27/95 |

## SAFETY EVALUATION FOR FEEDWATER ULTRASONIC TEST EQUIPMENT


### Summary:

This evaluation supports the temporary installation of ultrasonic flow test equipment to allow an assessment of feedwater venturi fouling. The test equipment used was a Caldon high precision ultrasonic flowmeter, which was mounted on the transducer mounting assembly. The transducer mounting assembly is clamped around the feedwater piping and connected to dedicated data collection equipment. The total duration of this temporary test setup was expected to last approximately one month.

Revision 2 specified the exact location of the feedwater piping for a calculation, which evaluated the seismic impact of the test equipment on the integrity of the feedwater piping.

Revision 3 allowed the mounting of the Caldon test equipment in locations different from that specified in previous revisions of this safety evaluation.


### Safety Evaluation:

This evaluation considered the seismic effects of the transducer mounting on the seismic qualifications of the feedwater piping and associated supports. With the transducer mounted around the outside of the piping, the pressure retaining capability of the piping was not affected. The data collection equipment had no potential for seismic interaction with any safety related equipment. No plant restrictions were identified, and this configuration did not affect Technical Specifications. Consequently, the special test equipment did not adversely affect any safety related equipment or functions and the proposed testing configuration 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 installation of test equipment or implementation of the testing identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-94-016
REVISION 2

UNIT : 3
APPROVAL DATE : 08/23/95

## SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF PRESSURE TRANSDUCERS FOR DP TESTING VARIOUS MOVs IN THE RHR, SI AND CCW SYSTEMS


### Summary:

This evaluation provided the basis for the acceptability of utilizing pressure transducers for the MOV differential pressure testing on several engineered safeguards systems. Testing was performed to demonstrate that the applicable MOVs develop sufficient thrust and torque to operate under maximum design basis differential pressure (DP) conditions. To perform the subject tests, pressure transducers were temporarily installed on the applicable portions of the SI, RHR and CCW systems. This testing was conducted to satisfy the requirements of NRC Generic Letter 89-10, Item c. Generic Letter 89-10 requires that the licensee develop and implement a comprehensive program for demonstrating and maintaining the operability of safety related MOVs and position-changeable MOVs. To perform the subject tests, pressure transducers were temporarily installed upstream and downstream of the MOVs to record actual system test pressures. This evaluation concluded that the method of implementation and limitations imposed by the test procedures with regard to the subject pressure transducers were consistent with all associated technical and licensing requirements.

Revision 2 involved an administrative change to the evaluation to allow for future testing of the identified MOVs in accordance with approved plant procedures without requiring additional revisions of the evaluation.


### Safety Evaluation:

The analyses, evaluations and implementation instructions supporting this activity ensured that no safety related systems, equipment or structure were adversely affected by the temporary installation of the pressure transducers. Temporary installation of the pressure transducers for testing purposes did not result in a change in the design function in any of the associated systems. No new failure modes were introduced and no adverse interactions with other equipment important to safety were introduced. The actions or 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 testing process identified within this evaluation.

**SAFETY EVALUATION JPN-PTN-SECJ-94-027**
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 02/09/95

## STAND ALONE SAFETY EVALUATION
## UNITS 3 AND 4 CONTAINMENT STRUCTURE ANALYSIS


**Summary:**

During the performance of the 20th year tendon surveillance of the Unit 3 and 4 containment structure post-tensioning systems, the measured normalized lift-off forces for a number of randomly selected tendons were found to be below the predicted lower limit (PLL). In accordance with plant Technical Specifications, the lift-off forces for the adjacent tendons were also examined and found to be below the predicted lower limit. As required by plant Technical Specifications engineering evaluations were prepared to address these lift-off conditions. These evaluations concluded that the most probable cause for the low measured lift-off forces was attributable to increased tendon wire relaxation loss due to average tendon wire temperatures higher than originally considered. These evaluations also recommended a structural re-analysis of the containment structure and post-tensioning system. The re-analysis of the Unit 3 and 4 containment and post-tensioning systems were documented in the above listed safety evaluation. This evaluation documents the minimum prestressing requirements utilizing the existing design and licensing bases as defined in Updated FSAR Appendix 5B. Attachment 1 to this evaluation contained a summary of the containment re-analysis methodology and results relative to the determination of new minimum prestressing requirements for each tendon group through the end of the licensed plant life, as well as an Update FSAR Change Package to document the affected areas of the Updated FSAR.


**Safety Evaluation:**

The containment structure and post-tensioning system is designed as a Safety Related Class I structure as defined in Appendix 5A of the Updated FSAR. The engineering re-analysis concluded that Turkey Point Units 3 and 4 containment post-tensioning systems will provide sufficient prestress force to maintain the existing licensing basis requirements beyond the licensed plant life. Consequently, the new minimum prestressing requirements did not have any adverse effect on plant safety or operation, 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 minimum prestressing requirements identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEIP-94-039
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 05/16/95

# SAFETY EVALUATION FOR CONTAINMENT AIRBORNE RADIATION MONITORS (R-11/R-12) ESF SETPOINTS

## Summary:

This evaluation documented the original design bases for the Containment Airborne Radiation Monitors R-11/R-12, documented how the setpoints used at Turkey Point were calculated, and demonstrated that currently implemented setpoints are conservative for the present operating conditions and for Technical Specifications. Also considered in this evaluation were the capabilities of R-11/R-12 to detect reactor coolant system (RCS) leakage (1 gpm leak) during normal operating modes and to detect high radiation releases from a fuel handling accident inside containment. The Containment Radiation Monitors R-11/R-12 measure airborne particulate and gaseous radioactivity levels within the containment building. The system provides an automatic signal to the Containment Ventilation Isolation and Control Room Ventilation Isolation systems in the event that preset radiation limits are exceeded. This evaluation concluded that the current ESF monitoring setpoints contained in Technical Specifications and their implementation are conservative considering current plant design, Technical Specifications requirements, and operating experience.

## Safety Evaluation:

This evaluation concluded that the Containment Airborne Radiation Monitoring system met its design bases with no changes required in plant Technical Specifications. Although changes were required to update the Updated FSAR to document previous licensing commitments, no changes were required to the setpoints currently used for either the R-11 or R-12 monitors or any other changes to plant equipment. Consequently, the existing plant setpoints and system condition 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 in the Updated FSAR identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEEP-94-041
REVISIONS 3 and 4

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| UNITS | : | | 3 & 4 |
| APPROVAL DATES | : | Rev.3 | 12/29/94 |
| | : | Rev.4 | 12/29/94 |

## SAFETY EVALUATION FOR OPERATION OF THE EMERGENCY LOAD SEQUENCERS WITH SOFTWARE LOGIC DEFECT IN TEST MODES


Summary:

This safety evaluation documented an assessment of a defect in the software logic for the emergency load sequencers detected during integrated safeguards testing. The emergency load sequencers are designed to load engineered safety features equipment onto the emergency power system during design basis events. The sequencers are programmable logic controllers (PLC), which receive inputs from various systems and function with timed logic operations to actuate equipment which is automatically started at specific intervals. During the Unit 4 integrated safeguards test, a failure of the 3A sequencer to respond to the opposite unit's Safety Injection (SI) signal occurred. Troubleshooting activities resulted in the discovery of a defect in the sequencer software logic which, under certain conditions, could inhibit the sequencer from responding to an emergency signal. This evaluation documented that the software logic defect was limited to sequencer test modes. Placing the sequencer test mode switch in the "OFF" position precluded the software defect from interfering with the safety function of the sequencers. Functional testing on the sequencer simulator using design basis inputs confirmed proper sequencer operation with the test mode switch placed in the "OFF" position.

Revisions 3 and 4 to this evaluation incorporated clarifications of the risk associated with manually testing (with the software defect) the sequencers on a monthly basis.


Safety Evaluation:

The operation of the sequencers with the auto-test feature disengaged did not adversely affect the operation of the sequencers as required for accident mitigation. This sequencer change reduced the potential for malfunctions of equipment important to safety. Also, disabling the auto-test feature did not introduce any new equipment failure modes. The actions and plant sequencer 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-SEMP-94-044**
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 12/29/94

## SAFETY EVALUATION FOR REVISION OF ULTIMATE HEAT SINK TECHNICAL SPECIFICATION BASES


### Summary:

This evaluation demonstrated that, with cooling canal temperatures of 100 °F or more, any two of three Component Cooling Water (CCW) heat exchangers are capable of satisfying their post-accident heat removal requirements. The cooling canals at Turkey Point are the ultimate heat sink for the Intake Cooling Water (ICW) system, which removes heat from the Component Cooling Water system for normal plant operation and during accident conditions. The plant's original Technical Specifications did not contain any requirements applicable to the plants ultimate heat sink. Later, a general upgrade of plant Technical Specifications included a requirement that the ultimate heat sink be operable with an average water temperature to the ICW system of less than or equal to 100 °F during operating Modes 1-4. This safety evaluation demonstrated that any two of three CCW heat exchangers can remove design basis accident heat loads with canal temperatures of greater than or equal to 100 °F; while actual documented heat exchanger performance will provide the basis for verifying the capability of the CCW heat exchangers to remove design basis accident heat loads with canal temperatures in excess of 100 °F. This safety evaluation was also developed to support a proposed revision of the Technical Specification Bases, which would provide additional clarifications for the upper limit of an acceptable ultimate heat sink temperature as well as provide clarifications for incorporation into the plant's Design Basis Documents.


### Safety Evaluation:

The activities addressed in this safety evaluation did not have any adverse effect on plant operations or plant safety. Similarly, the design basis clarifications addressed in this evaluation did not involve a change to the plant or to procedures described in Updated FSAR. Consequently, the analyses and design basis clarifications identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant Technical Specifications, other than an adminstrative change to the Bases. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SENS-94-044
REVISIONS 1 and 2

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| UNITS | : | | 3 & 4 |
| APPROVAL DATES | : | Rev.1 | 02/14/95 |
| | | Rev.2 | 10/03/95 |

## SAFETY EVALUATION FOR THE ABANDONMENT OF CETs AND HJTC SENSORS


### Summary:

This safety evaluation was developed to establish a basis for abandonment of several Core Exit Thermocouples (CETs) and Heated Junction Thermocouples (HJTC) which were discovered to be inoperable. The CETs and HJTCs are part of the Inadequate Core Cooling System (ICCS) instrumentation, which was designed to yield information on fuel assembly outlet temperatures at selected core locations. Using this information obtained from the ICCS instrumentation system, it would be possible to confirm the reactor core design parameters and calculate hot channel factors. The CETs are classified as Regulatory Guide 1.97 Type A Category 1 variables, while the HJTCs are classified as Regulatory Guide 1,97 Type B Category 1 variables. No repairs of the inoperable CETs/HJTCs were planned at that time, and the abandoned equipment procedure (0-ADM-220) was used to document the abandonment of the inoperable thermocouples. Regulatory Guide 1.97 states that Category 1 instrumentation channels should be available prior to an accident, except as specified within Technical Specifications. This evaluation documented that Technical Specification requirements for the number of remaining operable CETs/HJTCs continued to be satisfied without reliance on any abandoned instrumentation.

Revisions 1 and 2 document the abandonment of additional numbers of CET and/or HJTC sensors. The original conclusions of the safety evaluation remained unchanged.


### Safety Evaluation:

The ICCS system provides a means for acquiring information about the core, but does not perform any type of control function. Since the number of remaining operable CETs/HJCTs continued to satisfy Technical Specification requirements, there was no safety significance to the abandonment of several CETs/HJCTs. 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-SENP-94-045
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 11/18/94

## SAFETY EVALUATION FOR EXTENSION OF MANUAL TESTING INTERVAL FOR EMERGENCY LOAD SEQUENCERS


### Summary:

A one time extension of the manual testing interval for the emergency power system's load sequencers was addressed in this evaluation. A previous evaluation, JPN-PTN-SEEP-94-041, documented the discovery of a software logic error in the sequencer automatic and manual test modes. Evaluation JPN-PTN-SEEP-94-041 required that the sequencer test mode switch be placed in the "OFF" position to preclude the software error from interfering with its safety functions. JPN-PTN-SEEP-94-041 also confirmed the acceptability of a 30 day manual test frequency.

An extension of the sequencer manual test frequency to 6 a month interval was evaluated in this safety evaluation (JPN-PTN-SENP-94-045). Based on past and expected equipment reliability, this evaluation concluded that extending the time interval between sequencer manual tests was acceptable. In order to assess the risks of extending the sequencer manual test interval requirements, a probabilistic safety assessment was performed. The risks associated with extending the time interval between manual testing of the sequencers were insignificant based on the acceptance criteria of EPRI's draft PSA Applications Guide. Based on the above, this evaluation justified extending the sequencer manual test interval for up to 18 months. To be conservative, however, this one time extension between manual tests was limited to a six (6) month interval.

Revision 1 of this safety evaluation incorporated comments received from the Plant Nuclear Safety Committee (PNSC).


### Safety Evaluation:

This evaluation examined the load sequencer's design and licensing bases, Technical Specification requirements, failure modes, available functionality checks, and plant risk, and it concluded that it was acceptable to allow a one time extension of the sequencer test interval from one month to six months. Consequently, this evaluation concluded that extending the time interval between sequencer manual tests did not involve an unreviewed safety question nor did this proposed change require any 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 JPN-PTN-SEMS-94-051
REVISION 1

UNIT : 4
APPROVAL DATE : 03/21/96

# SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF PRESSURE TRANSDUCERS FOR DP TESTING VARIOUS MOVs IN THE RHR, SI AND CCW SYSTEMS

## Summary:

This safety evaluation addressed the testing that was performed on Residual Heat Removal (RHR), High Head Safety Injection (HHSI), and CCW system motor-operated valves (MOVs) to satisfy the valve testing criteria from NRC Generic Letter (GL) 89-10. NRC Generic Letter (GL) 89-10 requires that operating nuclear plants develop and implement programs which perform dynamic testing of safety related MOVs under conditions simulating maximum expected design basis conditions, including differential pressures. Testing was performed to demonstrate that the applicable MOVs develop sufficient thrust and torque to operate under the maximum design basis differential pressure conditions. To perform these differential pressure tests temporary pressure transducers were installed upstream and downstream of each valve, and the testing was performed while the unit was shutdown in operating Modes 5, 6, and defueled. This test equipment collected test data and monitored the performance of the MOV during actual differential pressure conditions. Most MOV testing was performed while the affected MOVs were out of service. However, for many valves, the MOV dynamic testing could only be performed during periods when the affected MOVs were required to be operable per existing plant procedures and/or Technical Specifications.

Revision 1 of this evaluation established the testing requirements for MOV-4-83A/B (HHSI) and those for administrative changes to allow for future testing of identified MOVs in accordance with approved procedures, without requiring additional revisions to this safety evaluation.

## Safety Evaluation:

This safety evaluation concluded that the method of implementation and limitations imposed by the test procedures with regard to the subject pressure transducers used for MOV testing are consistent with associated technical and licensing requirements. On this basis the safety evaluation demonstrated that the temporary installation of pressure transducers in the RHR, SI, and CCW systems to perform MOV differential pressure testing 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 JPN-PTN-SEMS-94-056
REVISION 0

UNIT : 4
APPROVAL DATE : 11/17/94

## SAFETY EVALUATION FOR INTAKE STRUCTURE PUMP OPERATION WITH STATIONARY SCREEN INSTALLED


### Summary:

This evaluation examined the potential safety impact of concurrent operation of an Intake Cooling Water (ICW) pump and a Circulating Water (CW) in a single intake bay with a stationary intake screen installed. Stationary screens were used to provide a debris barrier during periods when the travelling screens were out of service. After stationary screen installation, subsequent operating experience showed indications of ICW pump cavitation before loss of CW pump flow occurred. An engineering investigation determined that indications of ICW pump cavitation were actually due to air entrainment in the ICW flow, most probably from vortexing and/or a water fall across the stationary screen in the vicinity of the pump suction. This investigation concluded that the vortexing phenomenon occurred when the stationary screens had been clogged by large quantities of debris trapped against the screen, which created a significant change in water level across the screen (screen delta-pressure) and threatened the ICW pump by air entrainment. Under certain transient conditions, this evaluation also recognized that it may be desirable (to preserve operational flexibility) to run both a CW pump and an ICW pump in a single bay for short periods to accommodate pump switching evolutions. On this basis, the evaluation concluded that the concurrent operation of ICW and circulating water pumps from the same intake bay were acceptable only for short periods, such as may be necessary for pump switching and maintenance.


### Safety Evaluation:

This evaluation established a number of plant restrictions to ensure that the concurrent operation of an ICW pump and CW pump would not be aligned from the same intake bay, except as determined necessary by licensed operators to accommodate pump switching during transient operating conditions. Such periods would be limited to as short a time as possible, and an operator would continuously monitor the intake bay during such periods. On this basis the actions and operating restrictions established 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 operating restrictions identified in this evaluation.

## SAFETY EVALUATION JPN-PTN-SEIS-94-074
REVISION 0

UNIT : 4
APPROVAL DATE : 11/25/94

## SAFETY EVALUATION FOR TSA 4-94-49-9 UNIT 4 DELTA-T DEVIATION ALARM SETPOINT CHANGE


### Summary:

This evaluation was developed to support Temporary System Alteration (TSA) 4-94-49-9, which was implemented to make a temporary change in the Unit 4 Delta-T deviation alarm setpoint. Plant Technical Specifications establish operability requirements for various systems, such as, Overtemperature Delta-T and Overpower Delta-T, which utilize the Delta-T signal. However, the Delta-T deviation alarm setpoint change will not affect the Delta-T signals used for these and other plant systems. The alarm associated with this setpoint is used for control room annunciation only, and does not serve a safety related function. The Delta-T function develops a signal for the difference between the RCS hot and cold leg temperatures for each of the three RCS loops. The deviation alarm setpoint was changed from 3 °F to 4.5 °F. This change was evaluated by both FPL engineering and Westinghouse, and it was found to be an acceptable setpoint that would allieviate the nuisance alarms that had been experienced during a recent Unit 4 startup.


### Safety Evaluation:

This evaluation examined the difference between RCS loop C and the other two RCS loop Delta-T signals, which was causing control room annunciator actuations. Since the actual Delta-T signals were determined to be acceptable, the current alarms were considered to be nuisance alarms. The changed setpoints allowed a reasonable operating margin between the new operating range and the annunciator setpoint. Consequently, the temporary setpoint change 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 setpoint changes identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SECJ-95-001
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 08/10/95

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

### Summary:

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

### Safety Evaluation:

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

## SAFETY EVALUATION JPN-PTN-SEIS-95-001
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 05/09/95

## SAFETY EVALUATION FOR OPERABILITY OF EAGLE-21 RACKS WITH THE TEST SEQUENCE PROCESSOR (TSP) OUT OF SERVICE


### Summary:

This safety evaluation established a basis for the operability of the EAGLE-21 system with a Test Sequence Processor (TSP) disabled or out of service. The EAGLE-21 system is part of the Reactor Protection System (RPS) and is divided into five major subsystems: (1) power supply and distribution; (2) input/output; (3) loop processor; (4) testing; and (5) man-machine interface. The TSP is part of the testing subsystem. In the past, failure or unavailability the TSP feature had conservatively resulted in the plant declaring the associated EAGLE-21 rack out of service. During a review of the TSP function, it was confirmed that the TSP was not needed for the EAGLE-21 rack to perform its protective functions. Because all associated rack annunciators were locked-in due to the TSP being taken out of service, some additional surveillances were performed each shift by operating personnel as compensatory actions. Based on the above, this evaluation concluded that with the TSP feature inoperable, the plant protective functions provided by the Loop Processing Subsystem in the same EAGLE-21 rack continued to be fully functional and remained operable.


### Safety Evaluation:

This evaluation examined the design bases of the TSP subsystem and found that it did not affect the ability of the RPS to accommodate a single failure of a channel, the redundancy features of the RPS, and did not have any adverse interactions with other equipment important to safety. Based on the conclusions of this evaluation, the EAGLE-21 system with the test sequence processor inoperable continued to meet all design and licensing requirements and 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 JPN-PTN-SEMS-95-004
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 07/13/95

## SAFETY EVALUATION FOR A TEST OF THE USE OF ULTRAFINE FILTERS IN THE CVCS

### 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 level is reached. Phase III involves the permanent use of these filters under formal plant design change documentation. This evaluation only addressed Phase I 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 critieria 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-SEES-95-005
REVISION 2

UNIT : 3
APPROVAL DATE : 03/09/95

## SAFETY EVALUATION FOR THE UNIT 3 EDG START FAILURE RELAY RACE


### Summary:

During a rapid start test performed on Emergency Diesel Generator (EDG) 3B, the potential for a timed race between two EDG "start failure lockout" circuitry relays was identified after the 3B diesel experienced a "start failure lockout" malfunction. After an investigation of this condition, an operability assessment was performed and the above referenced 50.59 safety evaluation was developed to document the safety significance of the identified condition. This evaluation documented the effects of this relay race on the safety related design basis functions required for the EDGs. FPL discovered that the identified relay race in the start failure lockout circuitry is only applicable when a valid failure of an EDG to start has occurred. Therefore, this evaluation concluded that the malfunction of the EDG start failure lockout due to the identified relay race could not have adversely affected other required inputs to the lockout logic or prevented EDG control circuitry from starting an EDG, unless it had already failed to start for other reasons. This condition was also limited to Unit 3 and was not applicable to Unit 4, since the lockout logic for Unit 4 was different from Unit 3 (as the result of extensive emergency power modifications implemented in 1990-91) and not subject to the same type of start failure lockout malfunction. The potential for a start failure lockout malfunction will be corrected in an engineering plant change (PC/M) performed under 50.59 criteria.

Revision 2 of the evaluation added an evaluation under the criteria of 50.59, and incorporated adminstrative changes.


### Safety Evaluation:

This saftey evaluation addressed the safety consequences of an EDG start failure lockout and its effect on other EDG protective and control functions. The evaluation concluded that the start failure lockout malfunction could have only occurred after an EDG had failed to start for other reasons and would not have prevented an EDG from starting on demand. The failure of a single EDG to start is an analyzed failure mode in the design basis of the emergency power system. Consequently, the identified condition did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for operating the plant with this start failure lockout condition until plant modifications can be implemented.

SAFETY EVALUATION JPN-PTN-SENP-95-007
REVISION 1

UNIT : 3
APPROVAL DATE : 07/06/95

## 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 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 Residual Heat Removal (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 1 of this evaluation states that a proposed separation of the EDG test from the loss of offsite power test (part of ESIT testing) would be accomplished by the next Unit 3 refueling outage.


### 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-SEMP-95-008
REVISION 0

UNIT : 4
APPROVAL DATE : 03/07/95

## SAFETY EVALUATION FOR INSTALLATION OF TEMPORARY HANDWHEEL


### Summary:

This safety evaluation supported the use of Temporary System Alteration (TSA) 95-175, which installed a manual handwheel on the operator of valve LCV-4-460. LCV-4-460 is a 2-inch control valve located in the reactor coolant system (RCS) letdown line to the regenerative heat exchanger. It has no safety related functions other than to maintain the reactor coolant pressure boundary. LCV-4-460 also provides the capability to isolate letdown flow for an Appendix R fire from the Alternate Shutdown Panel. LCV-4-460 has a diaphragm operator which causes the valve to fail open on loss of instrument air. A manual handwheel was temporarily installed on the air operator of this valve to enable manual closure of the valve and to ensure that it was maintained closed while maintenance was performed on another letdown valve, CV-4-200A, located downstream of CV-4-460. The manual handwheel added to CV-4-460 was a standard option available for this valve and was of proven design and completely compatible with this valve.


### Safety Evaluation:

This evaluation examined the potential for new failure modes and any spatial interactions with other safety related equipment. The operation of valve CV-4-460 with the handwheel operator was administratively controlled using the plant's established clearance procedures. It's controlled closure as evaluated in the safety evaluation will ensure that double isolation for the RCS is maintained. Consequently the actions and plant restrictions established by the 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-95-011
REVISIONS 0 and 1

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|---|---|---|---|
| UNITS | : | | 3 & 4 |
| APPROVAL DATES | : | Rev.0 | 04/08/95 |
| | | Rev.1 | 04/26/95 |

## SAFETY EVALUATION FOR RELAXATION OF ICW FLOW VERIFICATION TO THE TPCW HEAT EXCHANGER DURING CCW BASKET STRAINER CLEANING


### Summary:

The purpose of this evaluation was to demonstrate that verification of Intake Cooling Water (ICW) flowrates to the Turbine Plant Cooling Water (TPCW) heat exchangers is not required during ICW/CCW basket strainer cleaning provided that two ICW pumps are considered to operate post-accident. The ICW operating procedures require operators to verify at least 8000 gpm ICW flow through the TPCW heat exchangers prior to taking an ICW header out of service for basket strainer cleaning. Situations have been identified where this flow could not be verified when cleaning of basket strainers would be necessary. The basis for this requirement was reviewed and a calculation was performed to demonstrate that with two ICW pumps operating, sufficient flow would be delivered to the Component Cooling Water (CCW) heat exchangers to remove design basis heat loads without having to account for ICW flow through the TPCW heat exchanger. Therefore, for the limited time when the plant takes an ICW header out of service for basket strainer cleaning, reliance on two ICW pumps in lieu of verifying ICW flow through the TPCW heat exchangers to meet ICW flow requirements to the CCW heat exchangers is acceptable.

Revision 1 extended the evaluation applicability to additional cases where a minimum 8000 gpm ICW flow is unavailable to the TPCW heat exchangers.


### Safety Evaluation:

This safety evaluation demonstrated that the operation of a second ICW pump (with three operable ICW pumps) is an acceptable alternative to assure sufficient flow to the CCW heat exchangers during periods when an ICW header is taken out of service. Therefore, the plant ICW operating procedures may be revised accordingly. Based on the calculations and assessment performed under this safety evaluation, the actions and changes in plant procedures 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 procedural changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMP-95-017
REVISION 0

UNIT : 3
APPROVAL DATE : 07/25/95

## SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF MOVATS TORQUE THRUST CELLS TO SUPPORT MOV DIAGNOSTIC TESTING


### Summary:

This safety evaluation addressed the impact of the temporary installation of test equipment on the operability of MOV-3-869, MOV-3-716B and MOV-3-1420. This evaluation also addressed the testing of MOV-3-869 and MOV-3-716B in Modes 5, 6 and defueled, and MOV-3-1420 during Modes 2 through 6 and defueled. NRC Generic Letter (GL) 89-10 requires that operating nuclear plants develop and implement programs to assure that safety related motor operated valves (MOVs) operate under maximum expected design basis conditions. Item c. of GL 89-10 requires that licensees perform dynamic MOV testing under conditions simulating maximum design basis conditions. In order to address GL 89-10 and document the results of the Item c. testing requirements, MOV diagnostic test equipment was installed on the valves and/or Limitorque actuators. This test equipment collected test data and monitored the performance of the MOV during actual dynamic testing. When test equipment was installed, the MOV testing was typically performed while the affected MOV and associated system are out of service. However, in selected instances, the MOV dynamic testing could only be performed during periods when the affected MOV and associated equipment are required to be operable per existing plant procedures and/or Technical Specifications.


### Safety Evaluation:

This safety evaluation addressed the temporary installation of the MOVATS torque thrust cells and associated operators and valves for structural capability, seismic qualification, and for valve operational requirements. The temporary installation of MOVATS torque thrust cells to facilitate MOV diagnostics did not adversely affect the operation of any system required to prevent or mitigate the consequences of design basis accidents and did not require any changes to plant Technical Specifications. Consequently, the temporary installation of MOVATS torque thrust cells 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 JPN-PTN-SEEP-95-021
REVISION 1

UNIT : 3
APPROVAL DATE : 08/23/95

# TEMPORARY DC BREAKER ASSIGNMENTS FOR NECESSARY LOADS DURING PC/M 95-021 IMPLEMENTATION

## Summary:


This safety evaluation was developed to support a Temporary System Alteration (TSA 3-95-003-22), which established alternative 125 VDC power to selected Load Center 3D01 circuits required during the implementation of PC/M 95-021. The Load Center 3D01 subpanel that contained obsolete molded case breakers, was replaced with a replacement panel containing new breakers as specified in PC/M 95-021. The safety evaluation addressed six 125 VDC circuits which were to remain energized with only a minimum interruption in service. During implementation of PC/M 95-021, the subject TSA temporarily powered the six circuits from spare breakers in the same DC Load Center in accordance with the requirements of the subject PC/M. After the replacement panel had been installed the six selected circuits were removed from their temporary power source and re-installed onto their permanently assigned breakers. The subject TSA was restricted in its implementation to the period in time after the reactor was defueled during the coincident outage of the 3A 4160 VAC bus. Various other restrictions were exercised to ensure that the requirements of the subject PC/M were properly implemented and emergency diesel loading requirements were protected.

Revision 1 was developed to expand the safety evaluation for the six selected circuits for the short period in time during which the six circuits would be without power during the transfer from their temporary source to their permanent source of power.


## Safety Evaluation:

As discussed in the safety evaluation, the temporary provisions and requirements for the six circuits were previously evaluated in PC/M 95-21. The TSA and associated safety evaluation were initiated to implement the requirements for providing temporary power to the selected circuits from spare breakers in the manner prescribed in PC/M 95-021. The evaluation demonstrated that during implementation of the subject TSA, the plant remained bounded by existing analyses and the requirements of Technical Specifications. Consequently, implementation of the subject TSA in support of PC/M 95-021 did not introduce an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the temporary actions or changes identified within this evaluation.

**SAFETY EVALUATION JPN-PTN-SEMS-95-023**
REVISION 2

UNITS : 3 & 4
APPROVAL DATE : 08/03/95

**SAFETY EVALUATION FOR PERFORMANCE OF MAIN STEAM SAFETY VALVE SETPOINT VERIFICATION TEST IN MODE 1**


**Summary:**

Main Steam Safety Valves (MSSVs) require periodic setpoint verification testing in accordance with Technical Specification 3/4.7.1 to demonstrate operability. This safety evaluation was issued to document an evaluation of the performance of the setpoint verification tests while in operating Modes 1, 2, or 3 and to examine the impact that the temporary Safety Valve Test (SVT) system had on the operability of the valves. Previously, Turkey Point verified the lift settings of the MSSVs while in Mode 3 with the tested valves declared out of service. In order to reduce critical path evolutions, a change in this testing approach was developed which would allow MSSV testing while in Modes 1, 2, or 3. Plant Technical Specifications require that all MSSVs remain operable in Modes 1-3. However, operation may continue with one or more MSSVs inoperable provided that within 4 hours: (1) either the valve is restored to an operable status; (2) the Power Range Neutron Flux High Trip Setpoint is reduced; or (3) power is reduced to below the specified rated thermal power level as required by Technical Specifications. This testing is performed utilizing a temporary Safety Valve Test system for determining the valve lift setting. This evaluation also examined testing while the valves remained in service. If a valve failed the testing acceptance criteria or fails to reseat, the MSSV would be declared inoperable and the applicable Technical Specification actions taken.

Revision 2 of this evaluation was issued to administratively incorporate comments that were identified during a Plant Nuclear Safety Committee meeting.


**Safety Evaluation:**

This evaluation examined the MSSV's design bases, the seismic aspects of testing, the effect of testing on MSSV operational requirements, failure modes and effects, plant operating restrictions during testing, and Technical Specification requirements. The evaluation concluded that the proposed testing approach and SVT equipment had no adverse impact on plant safety or plant operations, and therefore, did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SENP-95-023
REVISIONS 0 and 1

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| UNIT | : | | 4 |
| APPROVAL DATES | : | Rev.0 | 12/20/95 |
| | | Rev.1 | 03/27/96 |

## 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 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 Residual Heat Removal (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 1 addresses the temporary test configuration of each 4160 VAC bus during ESIT testing. 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-95-026
REVISIONS 0, 1, 2, and 3

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| UNITS | : | | 3 & 4 |
| APPROVAL DATES | : | Rev.0 | 09/19/95 |
| | | Rev.1 | 09/25/95 |
| | | Rev.2 | 09/28/95 |
| | | Rev.3 | 10/12/95 |

## SAFETY EVALUATION FOR CCW FLOW BALANCE AND POST-ACCIDENT ALIGNMENT REQUIREMENTS TO SUPPORT CURRENT AND UPRATED CONDITIONS

Summary:

During the process of performing Component Cooling Water (CCW) flow balancing analyses to meet the new requirements imposed by the FPL Thermal Uprate Project criteria, certain existing CCW flow configurations were identified for Unit 3 (shutdown in a refueling outage) which may have resulted in maximum CCW flows that could have potentially exceeded the flow limitations of certain CCW and safeguards components. This evaluation was developed to analyze these conditions and justify those procedure and engineering design changes that would preclude these conditions. This evaluation also examined operability issues and addressed the safety significance of these ICW flow concerns for both Units 3 and 4. As part of the overall set of corrective actions taken, a CCW system and component level Failure Modes and Effects Analysis was performed. This analysis supported the appropriateness and acceptability of corrective measures, comprised of procedural controls and plant design changes, to ensure that minimum CCW flows to Engineered Safety Features (ESF) equipment would be available post-accident while at the same time ensuring that maximum flow limits for CCW system components were observed. Based on the flow limits established in this safety analysis, the evaluation concluded that Unit 4 (operating unit) remained operable.

Revision 0 of this evaluation imposed restrictions on the plant which ensured the necessary controls to permit fuel reload (Mode 5 and 6 operation). Revisions 1 and 2 of this evaluation established additional requirements to permit Unit 3 operation up to and including MODE 1. Finally, Revision 3 revised the core damage frequencies provided in the risk analysis and evaluated the use of only one High Head SI pump to reduce CCW heat loads post-accident.

Safety Evaluation:

This safety evaluation concluded that the proposed corrective actions, comprised of procedural controls and design changes, would ensure that minimum CCW flows to engineered safety features equipment would be available post-accident while ensuring that maximum flow limits for CCW system components were observed. Consequently, the procedure and design 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 design changes and procedural changes identified within this evaluation.

**SAFETY EVALUATION JPN-PTN-SEMS-95-033**
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 08/10/95

## SAFETY EVALUATION FOR PENETRATION 65A HAWKE ROUND TRANSIT (HRT) SEAL

**Summary:**

This safety evaluation addressed the removal of the existing blind flanges on containment Penetration 65A (ILRT penetration) and the installation of a temporary spool piece with associated HRT seal on Units 3 and 4 during operating Modes 5, 6, or defueled. The temporary spool piece was installed on the exterior side of the penetration and contained all the electrical and hose connections, which were sealed within using an HRT seal. This temporary spool piece would be installed to facilitate running electrical cables and hoses required for the testing and the sludge lancing operation on the Steam Generators during outages. The spool piece met the requirements addressed in plant Procedure OSP-051.14 for temporary flanges to be installed on containment penetrations during refueling and would serve to prevent communication between the containment and outside atmosphere during periods of fuel movement.

**Safety Evaluation:**

This evaluation established plant restrictions and administrative controls for use of the temporary spool piece. It also evaluated the installed spool piece for dead loads, pressure requirements for reduced inventory operation, failure mechanisms, and found that this configuration would not adversely affect the safety of any refueling operation. Consequently, installation of this temporary spool piece did not introduce 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-SEFJ-95-034
REVISION 1

UNIT : 4
APPROVAL DATE : 11/07/95

## SAFETY EVALUATION OF SPENT FUEL POOL (SPF) BLACKNESS TESTING

**Summary:**

This safety evaluation was developed to establish the technical justification to support the performance of blackness testing of the Unit 4 spent fuel storage racks. Blackness testing is a technique used to measure the level of neutron absorption (degree of blackness) of the spent fuel racks with Boraflex or other neutron absorbing material(s) installed. High density spent fuel storage racks were installed in the spent fuel pool of Unit 4 in 1989. The design of the rack includes the use of Boraflex material, which is a strong neutron absorption material used to maintain the spent fuel in the pool in a subcritical array. Boraflex degradation has been an issue since 1987, and extensive industry study of the degradation of the Boraflex material has utilized the techniques associated with blackness testing and analysis of coupons. This safety evaluation addresses the blackness testing technique and the interaction of the test equipment with the spent fuel storage racks. This evaluation, performed under the criteria of 50.59, documents that the testing technique specified in the evaluation will not violate Technical Specification requirements for the spent fuel pool, will not violate heavy load requirements, and will not alter any margin of safety associated with the prevention and mitigation of fuel handling accidents.

Revision 1 of this evaluation added supplemental information to address Technical Specification requirements related to the margin of subcriticality that must be maintained for various testing configurations.

**Safety Evaluation:**

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

SAFETY EVALUATION JPN-PTN-SEMS-95-038
REVISION 0

UNIT : 4
APPROVAL DATE : 08/14/95

## SAFETY EVALUATION FOR THE USE OF FREEZE PLUGS TO SUPPORT IMPLEMENTATION OF PC/M 95-054


### Summary:

This safety evaluation addresses the use of freeze seals to temporarily isolate a section of Component Cooling Water (CCW) header. The freeze seals are required to install 1/2-inch tie-in connections that would permit aligning cooling water to the Post Accident Sample system (PASS) sample cooler from either Unit 3 or Unit 4 CCW systems. The freeze seals are necessary to isolate the tie-in locations from the operating system while the piping was open. The PASS cooler is normally cooled from Unit 3 CCW by a section of piping that cannot be isolated from those planned to supply chilled water to the Unit 3 Normal Containment Coolers (NCCs). However, the cold water supplied by the chillers would over-cool the PASS cooler. Therefore, the alternate supply and return lines were installed to allow the alignment of the PASS cooler to be independently supplied by the Unit 4 CCW. Since there were no valves to isolate the alternate supply and return tie-in locations from portions of the Unit 4 CCW system which would have to remain in service, two freeze seals were installed to provide isolation.


### Safety Evaluation:

The freeze seals were relied on to perform a CCW system boundary function only for a short period during the drilling and attachment of the tubing and associated isolation valve. Only one line was installed at a time. The strict controls imposed on the freeze seal process, the contingency measures, and the small size of the piping openings ensured that all CCW safety functions would remain unimpaired throughout the installation. Based on the precautions identified in this safety evaluation, FPL concluded that this modification 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 actions or changes identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SEMS-95-045
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 08/31/95

## REDUCED REACTOR COOLANT SYSTEM HYDROGEN CONCENTRATION

### Summary:

This evaluation addressed the requirements for a reduction in the concentrations of hydrogen present in the reactor coolant system (RCS) in preparation for a shutdown associated with an outage. During plant operation, the reactor coolant hydrogen concentration is maintained in the reactor coolant to control the radiolytic decomposition of water and other sources of oxygen ingress, since oxygen can accelerate RCS metal corrosion. However, a plant shutdown requires that reactor coolant hydrogen concentration be reduced below normal operating concentrations in preparation for plant refueling or where the RCS is opened to the atmosphere for maintenance. Previously degassing was accomplished after the unit was in Mode 3. To avoid the potential for adversely impacting an outage schedule, degassing the reactor coolant system prior to shutdown by reducing the hydrogen concentration to no less than 15 cc/kg while in Mode 1 was evaluated. The purpose of this safety evaluation was to demonstrate that the reduction in hydrogen concentration during Mode 1 did not involve an unreviewed safety question under NRC criteria. This safety evaluation concluded that a reduction in hydrogen concentration to no less than 15 cc/kg at power operation was acceptable, provided that the operating period did not exceed two days prior to shutdown and careful monitoring of the reactor coolant hydrogen was maintained.

### Safety Evaluation:

This evaluation examined shutdown chemistry requirements, potential for radiological and corrosion consequences, violation of procedural or Technical Specification requirements, limiting single failure consequences, LOCA and non-LOCA related analyses, containment analyses, and setpoints. Based on the evaluation performed, reducing the RCS hydrogen concentrations to no less than 15 cc/kg for no more than two days prior to shutdown did not involve an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SENS-95-052
REVISION 0

UNIT : 3
APPROVAL DATE : 09/30/95

# 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 FSAR 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. Upon loss of instrument air to the Unit 3 diesel fuel transfer valves, these valves fail closed which prevents the automatic transfer of fuel oil from the Unit 3 Diesel Oil Storage Tank to the EDG day tanks. Should the automatic fill isolation valves fail closed, operators have about 15 hours before the day tanks must be refilled to ensure continued operation of the Unit 3 EDGs. The use of 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. The Unit 4 EDG fuel transfer system was installed during 1991 modifications to the emergency power system and does not rely on the instrument air system for fuel transfer support functions. Therefore, 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.


## 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 Updated FSAR 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-SEES-95-058
REVISION 0

UNIT : 3
APPROVAL DATE : 10/26/95

## SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF A RECORDER ON THE 1BD ROD DRIVE POWER CABINET

### Summary:

During a Unit 3 startup a "Rod Control Urgent Failure" alarm was received in the control room for the Rod Control 1BD Power Cabinet. The Rod Control 1BD Power Cabinet contains the power circuits for Shutdown Bank B, Control Bank B, and Control Bank D Group 1 rods. An investigation revealed no apparent cause for the alarm. To continue the investigation of this event, the above referenced safety evaluation was developed to allow the temporary installation of an event recorder to monitor the various inputs to the rod control Urgent Failure alarm. The recorder was installed to monitor various signals in the rod control Power Cabinet 1BD and identify any abnormalities. This evaluation for the temporary installation of the recorder addresses the potential for both electrical and seismic interactions with other safety systems. The recorder was evaluated for overturning during a seismic event and did not represent a hazard for other safety systems.
An evaluation was also performed for electrical interactions, and the temporary monitoring installation would not introduce any new failure modes for the rod control system. 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 in rod control 1BD Power Cabinet would have no adverse impact on plant operations and would not have compromised the safety or licensing requirements for Unit 3. 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 JPN-PTN-SEMS-95-060
REVISION 0

UNIT : 3
APPROVAL DATE : 01/10/96

## SAFETY EVALUATION FOR THE TEMPORARY LOWERING OF UNIT 3 SPENT FUEL POOL (SFP) WATER LEVEL FOR MAINTENANCE ACTIVITIES

### Summary:

This evaluation was developed to examine the effects of securing the spent fuel cooling pumps and reduction in the pool level by about 2 feet in order to perform maintenance of the spent fuel pool (SFP) skimmer system. This evaluation addressed the effects of spent fuel handling accidents, spent fuel heatup rates; increased radiation levels resulting from lowered water (shielding) levels, and activation of system alarms. To reduce the potential for fuel handling accidents, all fuel movement and crane operation was suspended in accordance with Technical Specification 3/4.9.11. The spent fuel pool has been evaluated for elevated pool temperatures, and pool heatup from 100 °F to 125 °F was estimated to take about 8 hours, which would be a sufficient time to perform the required maintenance. Previous evaluations of reduced water levels have demonstrated that expected increases in radiation levels would be negligible. In order to preclude activation of the SFP alarms, the temperature of the pool was monitored every hour to detect water levels which would approach alarm levels, at which time work would be secured and water levels restored and cooling would be restored.

### Safety Evaluation:

This evaluation concluded that reducing the spent fuel pool level associated with maintenance on the pool skimmer will have no adverse impact on plant operations and will not compromise the spent fuel handling accident analyses, provided that the actions and restrictions identified in the evaluation are observed. Consequently, the reduced pool water level and other actions identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the actions or changes identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SECS-96-001
REVISION 0

UNIT : 4
APPROVAL DATE : 03/27/96

## EVALUATION FOR ALLOWING TOOLS AND EQUIPMENT TO REMAIN 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 operation. The items addressed under this evaluation, and the storage locations within Unit 4 containment were specifically identified within the evaluation. The purpose for 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 hydrogen generation within containment during accidents, the containment free volume and heat sink analyses, interactions with flow to the containment sump, and the potential for adverse interactions due to high energy line break jet impingement. 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.


### 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 operations, 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 JPN-PTN-SEFJ-96-002
REVISION 1

UNIT : 4
APPROVAL DATE : 02/14/96

## SAFETY EVALUATION TO INCREASE THE UNIT 4 CYCLE 15 EOC EXPOSURE AND TO INCLUDE A $T_{AVG}$/POWER COASTDOWN


### Summary:

The purpose of this safety evaluation was to allow Turkey Point Unit 4 to extend the design burnup of Cycle 15 from 13,523 MWD/MTU to 14,023 MWD/MTU. This was to be accomplished by using a $T_{avg}$ and power coastdown after reaching the end of reactivity at nominal hot full power conditions. The reactivity necessary to extend operations would be obtained from the negative moderator and power coefficients. Because these coefficients are negative, a decrease in either the moderator or the reactor power would result in a positive reactivity addition to the core, which would offset the reactivity loss from burnup. The $T_{avg}$ and power coastdown would be started at the end of Cycle 15 by reducing the full power primary $T_{avg}$ to a value no lower than 569.2 °F (5 °F below nominal) at a rate of approximately 1 °F per day. The power coastdown if needed would then be implemented by reducing the reactor power by approximately 1% per day. The core burnup would be limited to 14,023 MWD/MTU for the total coastdown.

Revision 1 of this safety evaluation specifically addressed the change from the Power Shape Sensitivity Model (PSSM) to an Explicit Shape Analysis for Peak Clad Temperature Effects (ESHAPE) in the LOCA analysis.


### Safety Evaluation:

This safety evaluation examined the effects of extending the EOC exposure and operating at reduced $T_{avg}$ and power using a standard, and proven Westinghouse methodology, and this evaluation determined that this would not have an adverse effect on plant safety or plant operations. Consequently, the plant actions and changes in plant procedures required to implement the approach 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 EOC exposure extension and $T_{avg}$/ power coastdown approach identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SEMS-96-003
REVISION 0

UNIT : 4
APPROVAL DATE : 02/29/96

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


### Summary:

This evaluation addressed the potential safety significance of operating the Unit 4 steam generators (SG) with foreign objects present in the secondary side. Foreign objects which are irretrievable have been identified within the secondary side of the Unit 4 steam generators. Previously, individual safety evaluations have addressed the acceptability of continued Unit 4 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 4 safety evaluation to assess and document all the Unit 4 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. FPL maintains a visual inspection program for the tube sheet annulus of the secondary side of steam generators (in addition to the other inspection programs for steam generators) to help prevent and detect the presence of loose parts during plant operation.


### Safety Evaluation:

Previous safety evaluations documented 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 current evaluation established 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 4 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 the presence of these foreign objects and the programmatic actions identified within this evaluation.

**SAFETY EVALUATION JPN-PTN-SEMP-96-004**
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 05/16/96

## 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.


**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 CWW piping identified within this safety evaluation.

## SAFETY EVALUATION JPN-PTN-SEMP-96-005
REVISION 0

UNIT : 4
APPROVAL DATE : 02/14/96

## SAFETY EVALUATION FOR INSTALLATION OF PRESSURE REGULATORS ON PORV ACTUATOR SUPPLY LINES

### Summary:

This safety evaluation was developed to support Plant Change Modification (PC/M) 95-164, which installed one regulator and one pressure indicator in each of the Instrument Air system supply lines to the power operated relief valves (PORVs) PCV-4-445C and PCV-4-456. In addition, to facilitate maintenance of the new regulators, the subject PC/M installed an isolation valve in the Instrument Air header upstream of the new regulators. During the performance of surveillance testing, the PORV stroke time is set by independently using the Instrument Air and Backup Nitrogen supply lines. These are independent supply lines that join into a common supply header upstream of the PORV supply accumulator. The PORV stroke time setting process is difficult and time consuming due to the differences in pressure between the two supply sources. As the result of installing regulators in each of the PORV instrument air supply lines, the PORV stroke time setting process can be improved, and the supply pressure to the PORV actuator diaphragm will be limited to the actuator design pressure of 100 psig. The permanent installation of regulators in the instrument air supply lines to the PORV actuators does not affect the PORV's design or safety functions. The Instrument Air system is a non-safety related system that has no required safety functions for the mitigation of design basis accidents. The pressurizer PORVs are also attached to the Backup Nitrogen supply which provides a safety related backup source for operation of the PORVs.

### Safety Evaluation:

The permanent installation of regulators in the instrument air supply lines to the PORV actuators does not affect the PORV's design requirements or functions. The Instrument Air system is a non-safety related system that has no required safety functions related to the pressurizer PORVs. This safety evaluation demonstrated that the activities involved in implementing PC/M 95-164 requirements did not have an adverse effect on plant safety or plant operation. Consequently, the installation of the instrument air regulators 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-96-007
REVISION 0

UNIT : 4
APPROVAL DATE : 02/20/96

## SAFETY EVALUATION FOR ICW VALVE REPAIR AND REPLACEMENT


### Summary:

This evaluation addressed the safety aspects for repair or replacement of five Intake Cooling Water (ICW) isolation valves during the 1996 Unit 4 refueling outage. In addition, three valves were removed from the ICW system and replaced with piping spool pieces. Plant Change Modification (PC/M) 94-131 documented the removal of valve CV-4-2202 and its replacement with a piping spool piece. Similarly, PC/M 95-090 documented the removal of CV-4-2201 and valve 4-50-403 and their replacement with a spool piece. This safety evaluation did not address activities associated with the above mentioned PC/Ms and was limited to an assessment of the repair and replacement of the initial five manual ICW valves mention above. Some of the valve replacements required one of the ICW headers to be removed from service. To ensure that Residual Heat Removal (RHR) and spent fuel pit (SFP) cooling requirements are met, ICW operations were controlled in accordance with the applicable Technical Specifications and system operating procedures. The ICW configurations that would be encountered during the valve repair and replacement were analyzed to ensure the continued ability of the ICW system to support RHR and SFP cooling requirements during Modes 5 and 6 or during the reactor defueled period.


### Safety Evaluation:

This evaluation addressed the temporary plant conditions and restrictions imposed on the operation of ICW while valves were out of service. The evaluation also addressed issues, such as, seismic effects, failure modes and effects, Technical Specification requirements, heavy loads, isolation boundaries, valve repair restrictions, and flow and heat removal capabilities. Because plant operating procedures and practices ensure decay heat removal functions are maintained, 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-SEES-96-008
REVISION 1

UNITS : 3 & 4
APPROVAL DATE : 02/10/96

## UNIT 4 CYCLE 16
## TEMPORARY CONTAINMENT POWER


**Summary:**

This evaluation addressed the effects of using existing spare cables and associated containment electrical penetrations for temporary power inside the Unit 4 containment during the Cycle 16 refueling outage. These cables and associated containment penetrations were abandoned in place (spared) when single speed Normal Containment Coolers (NCC) were placed into use. Prior refueling outages required that temporary power cables for containment work activities be routed through the open containment equipment hatch. However, several times during an outage, these cables would have to be removed so that the hatch could be closed for fuel movement and/or other containment integrity tests. Smaller capacity temporary power sources were set up during those periods. Without a source of power inside containment, almost all work would cease until these power cables could be re-routed back inside the containment. This was a time consuming process which slowed the progress of the refueling outage and affected overall plant availability. This safety evaluation was developed to support a Temporary System Alteration (TSA) for use of the NCC spare cables running through existing containment power penetrations to supply outage related power within containment.

Revision 1 of the evaluation changed the applicable plant operating modes to Mode 5 and 6, and it revised the expiration date of the evaluation to be consistent with the TSA.


**Safety Evaluation:**

This evaluation addressed the potential for seismic and electrical interactions and concluded that this refueling outage activity will have no adverse impact on plant operations, and would not compromise the safety and licensing bases for the Turkey Point plant. Consequently, the temporary use of the spare cables and associated containment penetrations during a refueling outage that were 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-96-009
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 02/18/96

## SAFETY EVALUATION RELATED TO MINIMUM ICW FLOW REQUIREMENTS DURING TEMPORARY SYSTEM REALIGNMENTS FOR STRAINER MAINTENANCE


### Summary:

This safety evaluation supports changes to plant procedures which clarify the precautions and limitations that are necessary to ensure that plant design bases for accident heat removal are met. During early 1996, cooling canal aquatic grass influx events occurred which could have potentially impacted the performance of the Intake Cooling Water (ICW) system. The ICW system basket strainers became obstructed, which resulted in periods of low ICW flow. Based on a conservative calculation using current ICW/CCW system design assumptions, up to 5 minutes of very low to no ICW flow to the Component Cooling Water (CCW) heat exchangers was found to be technically acceptable during ICW system realignments. Five minutes is also consistent with previous evaluations that have shown this time to be an acceptable time duration for a dedicated operator to take actions to realign the ICW system in the unlikely event an accident were to occur concurrently with the low flow event. On this basis applicable plant procedures were updated to incorporate the limiting time duration of up to 5 minutes, which allow for strainer valve manipulations and the restoration of full flow, during periods of low ICW flows.


### Safety Evaluation:

This safety evaluation provided a clarification of the design requirements for operation of the ICW system to allow for strainer valve manipulations in preparation for cleaning and the restoration of full flow to the ICW system without the need to unnecessarily enter into a Technical Specification ACTION statement. Based on a conservative calculation using current ICW/CCW system design assumptions, up to 5 minutes of very low to no ICW flow to the CCW heat exchangers was found to be acceptable during ICW system realignments. Consequently, the recommended procedural changes identified in this safety evaluation did not constitute an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for implementation of the procedural changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEIS-96-010
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 03/12/96

# SAFETY EVALUATION FOR ENVIRONMENTAL MONITORING EQUIPMENT SAFETY CLASSIFICATION


## Summary:

This evaluation documented the safety classification change of the Environmental Monitoring System, which is required to make estimates of offsite radiation exposure in the event of a gaseous radioactivity release. Meteorological data is collected to provide a measurement of three meteorological parameters required to make estimates of atmospheric dispersion: (1) wind speed; (2) wind direction; and (3) a measure of atmospheric stability. The operabilty of the environmental monitoring equipment was not addressed in the current plant Technical Speifications. This safety evaluation provided a basis for changing the safety classifcation of the Environmental Monitoring System from Quality Related to Not Nuclear Safety Related, for those portions of the Environmental Monitoring System which are not mounted in the control room.


## Safety Evaluation:

The proposed change in the safety clasification of the environmental monitoring system did not affect the ability of this system to fulfill the requirements identified in the Updated FSAR, Regulatory Guide 1.97, or plant Technical Specifications. Therefore, this change in safety classification 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 JPN-PTN-SENS-96-011
REVISION 0

UNITS : 3 & 4
APPROVAL DATE : 03/12/96

## SAFETY EVALUATION FOR THE USE OF MANUAL ACTIONS TO ISOLATE THE TPCW HEAT EXCHANGERS


**Summary:**

This evaluation established a basis for manual actions to isolate the non-safety related Turbine Plant Cooling Water (TPCW) system from the safety related Intake Cooling Water (ICW) system during periods when the normal automatic POV isolation are out of service for maintenance. The Pneumatically Operated Valves (POV) *-4882 and *-4883 isolate ICW flow to the two TPCW heat exchangers to ensure that sufficient flow is available to cool the safety related Component Cooling Water (CCW) heat exchangers when required for heat removal post-accident. Generic Letter 91-18 allows the use of manual actions in place of automatic features, if the physical differences between the automatic and manual actions are successfully evaluated. The evaluation addressed the ability of operators to recognize the signals requiring the manual action and operator access for the required manual action. Additionally, this evaluation addressed the written procedures and operator training (recommended in GL 91-18), which must be in place prior to making the substitution of manual actions for automatic features. This evaluation concluded that the use of operator actions to close manual ICW isolation valves to isolate the TPCW heat exchangers from ICW flow following a safety injection actuation signal (SIAS) or after a loss of offsite power (LOOP) was acceptable using a dedicated operator during the period when the POVs were out of service for maintenance. The required actions of the dedicated operator were proceduralized to ensure their successful execution.


**Safety Evaluation:**

This evaluation established an acceptable basis for the temporary substitution of manual operator actions to manually isolate ICW flow to the TPCW heat exchangers in place of the automatic POV isolation features during the period when the POV(s) would be out of service for maintenance. The evaluation demonstrated that this substitution would have no adverse effects on the safety related heat removal functions of either the ICW or CCW systems. Consequently, the substitution of manual actions for automatic features as discussed in this safety evaluation 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 actions identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SEMS-96-012
REVISION 0

UNIT : 4
APPROVAL DATE : 02/22/96

## SAFETY EVALUATION FOR FREEZE SEAL INSTALLATION TO SUPPORT MAINTENANCE ACTIVITIES ON 4-821


### Summary:

This safety evaluation addresses the use of freeze seals to temporarily isolate a section of the spent fuel pool (SFP) cooling header to perform maintenance activities on the primary water system fill connection to the SFP cooling system. The freeze seals were necessary to isolate the SFP cooling system to preclude the possibility of a partial drain-down of the SFP due to siphon action through the SFP cooling loop. This evaluation addressed issues associated with the potential for partial draining of the spent fuel pool due to freeze seal failures, and pool heat-up rates while the SFP cooling system was shutdown during maintenance. Freeze seals would be established and controlled by plant procedure, and contingency plans were established to minimize any pool leakage to the SFP equipment room to a level that could be handled by contaminated drains. To preclude the potential for fuel handling accidents, all fuel movement and crane operation was suspended in accordance with Technical Specification 3/4.9.11. The spent fuel pool was evaluated for elevated pool temperatures and pool heatup from 85 °F to 125 °F (an alarm setpoint) was estimated to take about 26 hours, which would be a sufficient time to restore cooling. Previous evaluations of reduced water levels have demonstrated that expected increases in radiation levels would be negligible. The temperature and water level of the pool were monitored every hour to verify water temperatures and levels which could approach alarm levels, at which time work would be secured and both water levels and cooling would be restored.


### Safety Evaluation:

This evaluation addressed various precautions to ensure the safe conduct of maintenance. To preclude the potential for a partial drain-down of the SFP due to loss of a freeze seal, the SFP cooling loop isolation valves were closed. Further precautions were implemented to impose strict controls on the freeze seal process and contingency measures were developed to ensure that pool levels remained above an alarm point. 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-SEMP-96-018
REVISION 0

UNITS : 4
APPROVAL DATE : 03/22/96

## SAFETY EVALUATION FOR THE TEMPORARY INSTALLATION OF MOVATS TORQUE THRUST CELLS TO SUPPORT MOV DIAGNOSTIC TESTING ON MOV-4-843A,B


### Summary:

This safety evaluation addressed the impact of the temporary installation of test equipment on the operability of MOV-4-843A, and MOV-4-843B. Since the high head saftey injection (HHSI) system is shared between Units 3 and 4, MOV-4-843A/B must remain operable to assure saftey injection flow path integrity for Unit 3 (Modes 1, 2, or 3) during the time that diagnostic test equipment will be installed. NRC Generic Letter (GL) 89-10 requires that operating nuclear plants develop and implement programs to assure that safety related motor operated valves (MOVs) operate under maximum expected design basis conditions. Item c. of GL 89-10 requires that licensees perform dynamic MOV testing under conditions simulating maximum design basis conditions. In order to address GL 89-10 and document the results of the Item c. testing requirements, MOV diagnostic test equipment was installed on the valves and/or Limitorque actuators. This test equipment collected test data and monitored the performance of the MOV during actual dynamic testing. When test equipment is installed, the MOV testing is typically performed while the affected MOV and associated system are out of service. However, for these two valves with Unit 3 in operating Modes 1, 2, or 3, the MOV dynamic testing could only be performed during a period when the affected MOV and associated equipment were required to be operable per existing plant procedures and/or Technical Specifications.


### Safety Evaluation:

This safety evaluation addressed the temporary installation of the MOVATS torque thrust cells and associated operators and valves for structural capability, seismic qualification, and for valve operational requirements. The temporary installation of MOVATS torque thrust cells to facilitate MOV diagnostics did not adversely affect the operation of any system required to prevent or mitigate the consequences of design basis accidents (Unit 3) and did not require any changes to plant Technical Specifications. Consequently, the temporary installation of MOVATS torque thrust cells 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 JPN-PTN-SEMP-96-019
REVISION 0

UNIT : 4
APPROVAL DATE : 03/27/96

# SAFETY EVALUATION FOR UNIT 4 CCW FLOW BALANCE REQUIREMENTS TO SUPPORT CURRENT AND UPRATED CONDITIONS


## Summary:

This evaluation documented the Component Cooling Water (CCW) flow analysis that was performed to establish the bounding flow cases for implementation of Thermal Uprate requirements. Florida Power and Light has initiated a program to increase the thermal power rating of the Nuclear Steam Supply (NSSS) system for Turkey Point from 2200 to 2300 MW(t). However, 2308 MW(t) was assumed for this evaluation. To enable implementation of the Thermal Uprate requirements between refueling outages without a unit shutdown, CCW flow restrictions were established to ensure that minimum CCW flows to Engineered Safety Features (ESF) equipment would be available post-accident while at the same time ensuring that maximum flow limits for the CCW system components were observed. In a previous evaluation of the CCW system to establish system flow balancing requirements, the CCW system components most limited by flow were the CCW heat exchangers. The CCW heat exchanger flow limitations ultimately dictated the maximum CCW system flow balancing restrictions for both units. The CCW heat exchangers for Unit 3 are older and smaller heat exchangers than those installed on Unit 4. For this reason the thermal power uprate analyses were revisited to determine if any additional CCW flow balancing margin could be obtained on Unit 4. This evaluation documents those revised CCW system flow balancing restrictions for Unit 4.


## Safety Evaluation:

In order to implement the conclusions of Unit 4 CCW system flow balance calculations, this safety evaluation demonstrated that the current plant design and licensing bases were preserved after the recommendations proposed by the Unit 4 CCW system flow balance calculations have been implemented. Similarly, this evaluation served to show that the proposed restrictions on CCW flow rates would not involve changes in plant Technical Specifications. Consequently, the proposed CCW flow balance 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 actions or changes identified within this evaluation.

SAFETY EVALUATION JPN-PTN-SENS-96-019
REVISION 0

UNIT : 4
APPROVAL DATE : 03/04/96

## SAFETY EVALUATION TO SUPPORT PRESSURE TESTING 4A RHR HEAT EXCHANGER


**Summary:**

This evaluation was developed to address pressure testing of the 4A Residual Heat Removal (RHR) heat exchanger. Traces of boric acid residue identified at the lower weep hole for the 4A RHR heat exchanger outlet nozzle were identified. The plant Chemistry department conducted a test of the residue and confirmed that it was boric acid. However, there was no observed leakage flow at the location, nor did the weep hole have any direct communication with the RHR system. In order to complete an operability assessment for this condition, additional data was required to confirm or deny the existence of the leak and to characterize the potential size of such a defect. That data was collected via a pressure test focused on identifying a leak rate through the weep hole. The pressure test was performed with the 4A RHR heat exchanger inoperable within a Technical Specification Action statement. This evaluation addressed the impact of supplying the suction of the pressure test pump from the 4A RHR pump discharge, which will be aligned to the Refueling Water Storage Tank (RWST), and the potential for radiation doses to the auxiliary building. This evaluation demonstrated that with manual actions to restore RHR system integrity, continued operability of the 4B RHR train could be assured with no impact on plant safety.


**Safety Evaluation:**

This evaluation concluded that this RHR system heat exchanger pressure test would not adversely affect the remaining operable RHR train provided that the requirements of this evaluation were met. The RHR test alignment did not adversely impact the RWST supply to the operable 4B RHR header, nor were post-accident radiation doses in the auxiliary building affected. Based on the precautions established in this evaluation, the pressure testing did not adversely impact plant safety or operation, did not constitute an unreviewed safety question, and did not require a change to Technical Specifications. Therefore, prior NRC approval was not required for activities associated with RHR 4A Heat Exchanger pressure test.

SAFETY EVALUATION JPN-PTN-SEMS-96-022
REVISION 0

UNIT : 4
APPROVAL DATE : 03/21/96

## SAFETY EVALUATION FOR PARTS OF A FLUORESCENT LAMP IN THE RCS


### Summary:

This evaluation examined the operation of the Unit 4 reactor coolant system (RCS) with the potential for parts from a broken fluorescent lamp present in the primary system. During the Unit 4 Cycle 16 refueling outage a 3-foot fluorescent work light tube was broken in the primary side hot leg plenum of the 4B steam generator. The tube from the work light was damaged when a nozzle cover landed on the light assembly. When the light was pulled from the steam generator manway, the rubber cover came off the end of the protective plastic cover dumping some of the contents into the steam generator. Actions were taken to retrieve the loose parts. The nozzle cover was placed over the hot leg nozzle to prevent any pieces from being knocked into the hot leg piping and reactor coolant system. The glass and fluorescent powder from the tube were vacuumed up and an inventory of the light parts was taken. All the parts were accounted for, however, the potential existed for some glass and powder to remain in the steam generator. The purpose of this evaluation was to assess the acceptability of operating Unit 4 with some of the chemical contents of the fluorescent tube and glass fragments potentially present in the reactor coolant system. This evaluation utilized loose parts evaluations performed on other primary systems.


### Safety Evaluation:

This evaluation examined the effects of both the fluorescent powder and glass fragments in the primary system, and it concluded that the chemical effects of the fluorescent material will not adversely affect the corrosion of the RCS or its chemistry control. Similarly, the presence of glass fragments in the primary system was not expected to cause any significant degradation in safety related equipment and would be expected to be reduced in size so that the glass particles could be filtered out of the primary fluid. Consequently, the presence of fluorescent powder and glass particles in the primary system did not involve an unreviewed safety question or require changes to the plant Technical Specifications. Therefore, prior NRC approval was not required for the conditions and actions identified within this evaluation.

## SAFETY EVALUATION JPN-PTN-SENS-96-024
REVISION 0

UNIT : 4
APPROVAL DATE : 03/21/96

## SAFETY EVALUATION TO SUPPORT 4A RHR PUMP AND 4B RHR HX REFUELING ALIGNMENT

### Summary:

This safety evaluation was developed to address the impact of a proposed Residual Heat Removal (RHR) system realignment on plant Technical Specification requirements for the RHR system 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 unavaiable due to an outage on the 4B 4160 VAC 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 Techncial 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.

### 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.

# SECTION 3

# RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 94-134

UNIT : 3
TURN OVER DATE : 10/13/95

## TURKEY POINT UNIT 3 CYCLE 15 RELOAD


### Summary:

This Engineering Package provided the reload core design for the Turkey Point Unit 3 Cycle 15 reload. The primary design change to the core for Cycle 15 was the replacement of 60 irradiated assemblies with 60 fresh Optimized Fuel Assembly (OFA) Region 17 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 fuel was arranged in a low leakage pattern with no significant differences between the Cycle 14 and Cycle 15 loading patterns. Region 17 used the same Debris Resistant Fuel Assemblies (DRFA) design as the prior Region 16, except that the grid spring force of the bottom grid was reduced to alleviate the potential for rod bow. This was the only mechanical design change for this reload. Cross core fuel bundle shuffles were utilized in the Cycle 15 loading pattern; these shuffles were adequate to minimize potential power asymmetries. However, a very slight eighth power asymmetry was predicted at BOC, resulting from a new loading pattern due to the mishandling of a twice-burned fuel assembly in the spent fuel pool. The asymmetry was the same in each quadrant, and no quadrant tilt was induced. In anticipation of a transition to the Revised Thermal Design Procedure (RTDP) methodology during mid-cycle operation, this package took into account both the Standard Thermal Design Procedure (STDP) and RTDP. FPL expects to uprate the unit to a maximum thermal power of 2300 MW(th) early in the cycle; however, a new PC/M will be required to address this uprate after NRC approval has been obtained.


### Safety Evaluation:

The Unit 3 Cycle 15 reload core design was evaluated by FPL and by the fuel supplier, Westinghouse Electric. The Cycle 15 reload core design met all applicable design criteria, appropriate licensing bases, and the requirements of plant Technical Specifications. Any minor design modifications to fuel assemblies and core components 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 15 core reload did not have any adverse effect on plant safety or plant operations. Consequently, the Cycle 15 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 15 core reload.

## PLANT CHANGE/MODIFICATION 95-066

UNIT : 4
TURN OVER DATE : 05/20/96

## TURKEY POINT UNIT 4 CYCLE 16 RELOAD

### Summary:

This Engineering Package provided the reload core design for the Turkey Point Unit 4 Cycle 16 reload. The primary design change to the core for Cycle 16 was the replacement of 64 irradiated assemblies with 64 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 fuel was arranged in a low leakage pattern with no significant differences between the Cycle 15 and Cycle 16 loading patterns. Region 18 used the same Debris Resistant Fuel Assemblies (DRFA) design as the prior Region 17 fuel. There were no mechanical design changes for this reload. Cross core fuel bundle shuffles were utilized in the Cycle 16 loading pattern; these shuffles were adequate to minimize potential power asymmetries. In anticipation of a transition to the Revised Thermal Design Procedure (RTDP) methodology during mid-cycle operation, this package took into account both the Standard Thermal Design Procedure (STDP) and RTDP. FPL expects to uprate the unit to a maximum thermal power of 2300 MW(th) early in the cycle; however, a new PC/M will be required to address this uprate after NRC approval has been obtained. This engineering package evaluated a full core off-load to the spent fuel pool and revised the Updated FSAR: (1) for this reload cycle; and (2) to better reflect commitments to the NRC related to the high density spent fuel storage racks.

### Safety Evaluation:

The design of the Unit 4 Cycle 16 reload core was evaluated by FPL and by the fuel supplier, Westinghouse Electric. The Cycle 16 reload core design met all applicable design criteria, all pertinent licensing bases, and the requirements of plant Technical Specifications. Any minor design modifications to fuel assemblies and core components did not affect applicable design criteria for these components 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 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 its 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).

No power operated relief valve (PORV) challenges occurred for the Turkey Point Plant Units 3 and 4 during the period from November 14, 1994 through April 7, 1996.

### Unit 3

None.

### Unit 4

None.

SECTION 5

STEAM GENERATOR TUBE INSPECTIONS FOR TURKEY POINT

FORM NIS-BB OWNERS' DATA REPORT FOR EDDY CURRENT EXAMINATION RESULTS

As required by the provisions of the ASME CODE RULES

EDDY CURRENT EXAMINATION RESULTS

PLANT : Turkey Point Unit 3

EXAMINATION DATE: SEPTEMBER 14, 1995 thru SEPTEMBER 19, 1995

| 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% - 100% | | | |
| 3E210A | 3197 | 20 | NONE | NONE | NONE | 17 |
| 3E210B | 3198 | 35 | 2 | NONE | 2 | 18 |
| 3E210C | 3181 | 30 | NONE | NONE | NONE | 33 |

LOCATION OF INDICATIONS

(20% - 100%)

| 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% TO 100% |
| 3E210A | 5 | 4 | 6 | 1 | NONE | 5 | 21 | NONE |
| 3E210B | 8 | 10 | 8 | 6 | 3 | 8 | 41 | 2 |
| 3E210C | 33 | 3 | 1 | 9 | 1 | NONE | 47 | NONE |

Remarks:

DATE: 11/15/95 PREPARED BY: Alonso Montalbano G.
S/G EDDY CURRENT COORDINATOR

DATE: 11/15/95 REVIEWED BY: [signature]
INSPECTIONS SUPERVISOR

DATE: 11/16/95 REVIEWED BY: Gary L Borders
S/G TECHNICAL SPECIALIST

forms &/nisbb/p3 drw

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/95

COMPONENT : S/G A

Page : 1
Date : 11/15/95

Examination Dates : 09/14/95 thru 09/19/95

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

Total Indications
Between 20% and 39% .............: 21
Greater than or equal to 40% ...: 0

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

Location Of Indications 20% to 100%

| Hot Leg | | Cold Leg | |
|---|---|---|---|
| TSH -.5 to 01H -2.1 : | 5 | TSC -.5 to 01C -2.1 : | 0 |
| 01H -2.0 to 06H +2.0 : | 6 | 01C -2.0 to 06C +2.0 : | 4 |
| 06H +2.1 to AV1 -3.1 : | 1 | 06C +2.1 to AV4 -3.1 : | 0 |
| AV1 -3.0 to AV4 -3.0 : | 5 | | |

# INDICATIONS/TRENDING REPORT

PTN-3

OUTAGE : 09/95

COMPONENT : S/G A

DESCRIPTION : 20% to 39%

Page : 1

Date : 10/ 3/95

Time : 15:02:59

| | | | Extent | | | | 09/95 | | | | | | N/A | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Row | Col | Leg | Req | Tst/Note | Reel | Probe | Location | Volts | Deg | Ch | % | Diff | Location | Volts | Deg | Ch | % |
| 13 | 5 | C | TEH | TEH | AC008 | A-720-M/ULC | TSH 3.8 | .8 | 151 | 1 | 27 | | | | | | |
| 14 | 7 | C | TEH | TEH PS | AC007 | A-720-M/ULC | TSH 3.9 | .3 | 152 | 1 | 27 | | | | | | |
| 12 | 9 | C | TEH | TEH SS | AC008 | A-720-M/ULC | 04C 47.9 | .3 | 153 | 1 | 22 | | | | | | |
| 33 | 15 | C | TEH | TEH SS | AC016 | A-720-M/ULC | TSH .6 | .5 | 94 | 1 | 29 | | | | | | |
| 4 | 20 | C | TEH | TEH SS | AC011 | A-720-M/ULC | 01C 32.4 | .3 | 149 | 1 | 29 | | | | | | |
| 40 | 27 | C | TEH | TEH PS | AC023 | A-720-M/ULC | 04H 6.1 | .3 | 151 | 1 | 28 | | | | | | |
| 33 | 40 | C | TEH | TEH GNPL | AC020 | A-720-M/ULC | 05C 2.4 | .4 | 139 | 1 | 34 | | | | | | |
| 33 | 41 | C | TEH | TEH PS | AC020 | A-720-M/ULC | AV1 .0 | .8 | | P 2 | 25 | | | | | | |
| 38 | 45 | C | TEH | TEH PS | AC019 | A-720-M/ULC | AV2 .0 | 1.5 | | P 2 | 33 | | | | | | |
| | | C | TEH | TEH PS | AC019 | A-720-M/ULC | AV3 .0 | .7 | | P 2 | 25 | | | | | | |
| 37 | 47 | C | TEH | TEH PS | AC045 | A-720-M/ULC | AV3 .0 | .6 | | P 2 | 24 | | | | | | |
| 22 | 52 | C | TEH | TEH | AC038 | A-720-M/ULC | 04H 40.6 | 1.3 | 139 | 1 | 37 | | | | | | |
| 28 | 59 | C | TEH | TEH PS | AC044 | A-720-M/ULC | AV2 .0 | .6 | | P 2 | 22 | | | | | | |
| 38 | 65 | C | TEH | TEH | AC047 | A-720-M/ULC | AV3 .0 | .4 | | P 2 | 20 | | | | | | |
| 20 | 68 | C | TEH | TEH | AC041 | A-720-M/ULC | 04H 42.2 | .4 | 151 | 1 | 29 | | | | | | |
| 30 | 69 | C | TEH | TEH PS | AC044 | A-720-M/ULC | 03H 1.4 | .5 | 146 | 1 | 28 | | | | | | |
| 10 | 75 | C | TEH | TEH PS | AC053 | A-720-M/ULC | 04H 44.7 | .4 | 139 | 1 | 35 | | | | | | |
| 9 | 76 | C | TEH | TEH PS | AC052 | A-720-M/ULC | 02C 16.4 | 1.4 | 146 | 1 | 27 | | | | | | |
| 33 | 78 | C | TEH | TEH PC | AC047 | A-720-M/ULC | TSH .7 | .6 | 151 | 1 | 28 | | | | | | |
| 18 | 81 | C | TEH | TEH PS | AC049 | A-720-M/ULC | 01H 4.9 | .2 | 135 | 1 | 39 | | | | | | |
| 22 | 86 | C | TEH | TEH PS | AC049 | A-720-M/ULC | BAH 12.2 | .3 | 145 | 1 | 30 | | | | | | |

Number of RECORDS Selected from Current Outage : 21

Number of TUBES Selected from Current Outage : 20

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/95

COMPONENT : S/G B

Page : 1
Date : 11/15/95

Examination Dates : 09/14/95 thru 09/19/95

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

Total Indications
Between 20% and 39% .............: 41
Greater than or equal to 40% ...: 2

Total Tubes Plugged as Preventive Maint : 0
Total Tubes Plugged ....................: 2

Location Of Indications 20% to 100%

| Hot Leg | | Cold Leg | |
|---|---|---|---|
| TSH -.5 to 01H -2.1 : | 8 | TSC -.5 to 01C -2.1 : | 3 |
| 01H -2.0 to 06H +2.0 : | 8 | 01C -2.0 to 06C +2.0 : | 10 |
| 06H +2.1 to AV1 -3.1 : | 4 | 06C +2.1 to AV4 -3.1 : | 2 |
| AV1 -3.0 to AV4 -3.0 : | 8 | | |

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 09/95

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

Page : 1
Date : 10/ 3/95
Time : 15:17:24

| | | | Extent | | | | | 09/95 | | | | | N/A | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Row | Col | Leg | Req | Tst/Note | Reel | Probe | Location | Volts | Deg | Ch | % | Diff | Location | Volts | Deg | Ch | % |
| 11 | 3 | H | TEC | TEC GWPL | BH017 | A-720-M/ULC | 02C 36.1 | .4 | 150 | 1 | 36 | | | | | | |
| 10 | 4 | H | TEC | TEC PS | BH012 | A-720-M/ULC | 02C 32.0 | .3 | 148 | 1 | 34 | | | | | | |
| 10 | 7 | H | TEC | TEC | BH012 | A-720-M/ULC | 02C 31.8 | .7 | 157 | 1 | 27 | | | | | | |
| 5 | 8 | H | TEC | TEC | BH012 | A-720-M/ULC | 04H 15.0 | 2.0 | 148 | 1 | 34 | | | | | | |
| 23 | 9 | C | TEH | TEH | BC042 | A-720-M/ULC | 05C 31.1 | .6 | 144 | 1 | 35 | | | | | | |
| | | C | TEH | TEH SS | BC042 | A-720-M/ULC | 05C 7.7 | .3 | 148 | 1 | 31 | | | | | | |
| 11 | 10 | H | TEC | TEC | BH018 | A-720-M/ULC | 02C 36.6 | .6 | 151 | 1 | 30 | | | | | | |
| 23 | 10 | C | TEH | TEH PC | BC037 | A-720-M/ULC | 06H 4.0 | 1.3 | 150 | 1 | 31 | | | | | | |
| 10 | 11 | H | TEC | TEC PL | BH012 | A-720-M/ULC | 02C 32.0 | .4 | 152 | 1 | 31 | | | | | | |
| 11 | 11 | H | TEC | TEC PS | BH018 | A-720-M/ULC | 02C 36.8 | .5 | 155 | 1 | 26 | | | | | | |
| 18 | 12 | H | TEC | TEC PS | BH018 | A-720-M/ULC | 05H 20.0 | .5 | 158 | 1 | 24 | | | | | | |
| 26 | 20 | C | TEH | TEH PS | BC037 | A-720-M/ULC | AV4 .0 | .4 | | P 2 | 23 | | | | | | |
| 40 | 25 | H | TEC | TEC | BH020 | A-720-M/ULC | 05C 38.8 | .6 | 159 | .1 | 28 | | | | | | |
| 32 | 34 | H | TEC | TEH PS | BC004 | A-720-M/ULC | AV1 .0 | .5 | | P 2 | 27 | | | | | | |
| | | H | TEC | TEH PS | BC004 | A-720-M/ULC | AV3 .0 | .9 | | P 2 | 30 | | | | | | |
| | | H | TEC | TEH PS | BC004 | A-720-M/ULC | AV4 .0 | .5 | | P·2 | 27 | | | | | | |
| 11 | 38 | C | TEH | TEH SC | BC030 | A-720-M/ULC | 01H 12.2 | .4 | 145 | 1 | 29 | | | | | | |
| 34 | 38 | H | TEC | TEC SC | BH019 | A-720-M/ULC | AV1 .0 | .4 | | P 2 | 22 | | | | | | |
| | | H | TEC | TEC | BH019 | A-720-M/ULC | AV2 .0 | 2.6 | | P 2 | 38 | | | | | | |
| 42 | 38 | C | TEH | TEH PS | BC004 | A-720-M/ULC | TSH 1.5 | 1.2 | 142 | 1 | 36 | | | | | | |
| | | C | TEH | TEH PS | BC004 | A-720-M/ULC | TSH 3.5 | .6 | 151 | 1 | 25 | | | | | | |
| 11 | 39 | C | TEH | TEH | BC029 | A-720-M/ULC | 01H 12.2 | .7 | 156 | 1 | 26 | | | | | | |
| 30 | 42 | C | TEH | TEH PS | BC029 | A-720-M/ULC | AV3 .0 | .4 | | P 2 | 21 | | | | | | |
| 21 | 43 | C | TEH | TEH PS | BC030 | A-720-M/ULC | 04C 49.5 | .8 | 142 | 1 | 33 | | | | | | |
| 6 | 44 | C | TEH | TSH PS | BC031 | A-720-M/ULC | TSH 39.3 | 1.6 | 148 | 1 | 31 | | | | | | |
| 34 | 46 | H | TEC | TEC | BH020 | A-720-M/ULC | AV2 .0 | 1.2 | | P 2 | 29 | | | | | | |
| | | H | TEC | TEC | BH020 | A-720-M/ULC | AV3 .0 | 2.0 | | P 2 | 36 | | | | | | |
| 34 | 51 | H | TEC | TEC PS | BH021 | A-720-M/ULC | AV2 .0 | 1.7 | | P 2 | 31 | | | | | | |
| 45 | 52 | H | TEC | TEC | BH021 | A-720-M/ULC | TSH .7 | 1.7 | 162 | 1 | 24 | | | | | | |
| 42 | 55 | H | TEC | TEC PC | BH020a | A-720-M/ULC | AV3 .0 | 1.9 | | P 2 | 35 | | | | | | |
| 44 | 57 | H | TEC | 05H PS | BH020a | A-720-M/ULC | TSH .7 | 2.0 | 166 | 1 | 21 | | | | | | |
| 37 | 59 | H | TEC | TEH PC | BC013 | A-720-M/ULC | 05H 7.0 | .4 | 159 | 1 | 24 | | | | | | |
| 43 | 60 | H | TEC | 05H PS | BH021 | A-720-M/ULC | TSH .7 | 1.5 | 158 | 1 | 28 | | | | | | |
| 27 | 70 | C | TEH | TEH PS | BC035 | A-720-M/ULC | 06H 4.4 | .6 | 151 | 1 | 30 | | | | | | |
| 20 | 81 | C | TEH | TEH RS | BC035 | A-720-M/ULC | BAC .7 | .7 | 134 | P 1 | 25 | | | | | | |
| 29 | 81 | C | TEH | TEH SS | BC036 | A-720-M/ULC | 05H 15.8 | .5 | 144 | 1 | 30 | | | | | | |
| 23 | 86 | C | TEH | TEH SS | BC036 | A-720-M/ULC | TSC .7 | 1.8 | 80 | 1 | 35 | | | | | | |
| 11 | 89 | C | TEH | TEH PS | BC035 | A-720-M/ULC | 02H 36.4 | .6 | 150 | 1 | 30 | | | | | | |
| 16 | 89 | C | TEH | TEH PS | BC035 | A-720-M/ULC | TSC .7 | 1.0 | 128 | P 1 | 25 | | | | | | |
| 11 | 90 | C | TEH | TEH PS | BC036 | A-720-M/ULC | 02H 36.5 | .4 | 146 | 1 | 32 | | | | | | |
| 7 | 92 | H | TEC | TEC PL | BH015 | A-720-M/ULC | TSH .7 | .9 | 152 | 1 | 31 | | | | | | |

Number of RECORDS Selected from Current Outage : 41

Number of TUBES Selected from Current Outage : 35

INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 09/95

COMPONENT : S/G B
ESCRIPTION : 40% to 100%

Page : 1
Date : 10/ 3/95
Time : 15:17:56

| | | | Extent | | | | 09/95 | | | | | | N/A | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Row | Col | Leg | Req | Tst/Note | Reel | Probe | Location | Volts | Deg | Ch | % | Diff | Location | Volts | Deg | Ch | % |
| 42 | 37 | C | TEH | TEH | BH026 | MRPC680-3C | TSH .7 | 1.1 | 134 | CHR | 44 | | | | | | |
| 34 | 38 | H | TEC | TEC PL | BH019 | A-720-M/ULC | AV3 .0 | 3.3 | | P 2 | 42 | | | | | | |

Number of RECORDS Selected from Current Outage : 2

Number of TUBES Selected from Current Outage : 2

CUMULATIVE DISTRIBUTION SUMMARY
TURKEY POINT UNIT # 3
09/95

COMPONENT : S/G C

Page : 1
Date : 11/15/95

Examination Dates : 09/14/95 thru 09/19/95

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

Total Indications
Between 20% and 39% ............: 47
Greater than or equal to 40% ...: 0

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

Location Of Indications 20% to 100%

| Hot Leg | | Cold Leg | |
|---|---|---|---|
| TSH -.5 to 01H -2.1 : | 0 | TSC -.5 to 01C -2.1 : | 1 |
| 01H -2.0 to 06H +2.0 : | 1 | 01C -2.0 to 06C +2.0 : | 3 |
| 06H +2.1 to AV1 -3.1 : | 3 | 06C +2.1 to AV4 -3.1 : | 6 |
| AV1 -3.0 to AV4 -3.0 : | 33 | | |

# INDICATIONS/TRENDING REPORT
PTN-3
OUTAGE : 09/95

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

Page : 1
Date : 10/ 3/95
Time : 15:20:51

| | | | Extent | | | | 09/95 | | | | | | N/A | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Row | Col | Leg | Req | Tst/Note | Reel | Probe | Location | Volts | Deg | Ch | % | Diff | Location | Volts | Deg | Ch | % |
| 3 | 10 | N | 06H | 06H | CH004 | A-720-M/ULC | 02H 34.8 | .9 | 146 | 1 | 35 | | | | | | |
| 16 | 15 | C | TEN | TEN PC | CC039 | A-720-M/ULC | 06C 1.6 | .5 | 156 | 1 | 28 | | | | | | |
| 31 | 15 | C | TEN | TEN PS | CC036 | A-720-M/ULC | AV1 .0 | .3 | | P 2 | 20 | | | | | | |
| 40 | 25 | C | TEN | TEN SS | CC033 | A-720-M/ULC | AV2 .0 | .5 | | P 2 | 21 | | | | | | |
| | | C | TEN | TEN SS | CC033 | A-720-M/ULC | AV3 .0 | .6 | | P 2 | 22 | | | | | | |
| 37 | 28 | C | TEN | TEN PS | CC034 | A-720-M/ULC | AV4 .0 | .4 | | P 2 | 20 | | | | | | |
| 27 | 30 | C | TEN | TEN PS | CC032 | A-720-M/ULC | AV3 8.2 | 1.7 | 152 | 1 | 20 | | | | | | |
| 30 | 31 | C | TEN | TEN PS | CC029 | A-720-M/ULC | AV2 .0 | .3 | | P 2 | 20 | | | | | | |
| | | C | TEN | TEN PS | CC029 | A-720-M/ULC | AV3 .0 | .5 | | P 2 | 25 | | | | | | |
| 33 | 31 | C | TEN | TEN PS | CC031 | A-720-M/ULC | AV3 .0 | .5 | | P 2 | 26 | | | | | | |
| 34 | 31 | C | TEN | TEN PL | CC031 | A-720-M/ULC | AV2 .0 | .5 | | P 2 | 26 | | | | | | |
| | | C | TEN | TEN PL | CC031 | A-720-M/ULC | AV3 .0 | .3 | | P 2 | 24 | | | | | | |
| 4 | 33 | C | TEN | TEN SS | CC030 | A-720-M/ULC | TSC 28.5 | .5 | 149 | 1 | 24 | | | | | | |
| 43 | 33 | C | TEN | TEN PC | CC036 | A-720-M/ULC | AV3 .0 | .3 | | P 2 | 20 | | | | | | |
| 35 | 35 | C | TEN | TEN PS | CC031 | A-720-M/ULC | AV3 .0 | .4 | | P 2 | 24 | | | | | | |
| 35 | 36 | C | TEN | TEN PC | CC034 | A-720-M/ULC | AV3 .0 | .2 | | P 2 | 20 | | | | | | |
| 34 | 41 | C | TEN | TEN PS | CC031 | A-720-M/ULC | AV1 .0 | .8 | | P 2 | 32 | | | | | | |
| | | C | TEN | TEN PS | CC031 | A-720-M/ULC | AV3 .0 | .5 | | P 2 | 27 | | | | | | |
| | | C | TEN | TEN PS | CC031 | A-720-M/ULC | AV4 .0 | .8 | | P 2 | 31 | | | | | | |
| 33 | 43 | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV2 .0 | .4 | | P 2 | 22 | | | | | | |
| | | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV3 .0 | .4 | | P 2 | 22 | | | | | | |
| 35 | 43 | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV1 .0 | .3 | | P 2 | 21 | | | | | | |
| | | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV2 .0 | .6 | | P 2 | 28 | | | | | | |
| | | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV3 .0 | 1.0 | | P 2 | 34 | | | | | | |
| | | C | TEN | TEN PS | CC033 | A-720-M/ULC | AV4 .0 | .7 | | P 2 | 29 | | | | | | |
| 35 | 44 | C | TEN | TEN | CC036 | A-720-M/ULC | AV2 .0 | 1.1 | | P 2 | 31 | | | | | | |
| | | C | TEN | TEN | CC036 | A-720-M/ULC | AV3 .0 | 1.1 | | P 2 | 32 | | | | | | |
| | | C | TEN | TEN | CC036 | A-720-M/ULC | AV4 .0 | .5 | | P 2 | 24 | | | | | | |
| 30 | 48 | C | TEN | TEN PL | CC018 | A-720-M/ULC | AV2 .0 | .8 | | P 2 | 28 | | | | | | |
| | | C | TEN | TEN PL | CC018 | A-720-M/ULC | AV3 -.1 | 1.4 | | P 2 | 36 | | | | | | |
| 35 | 49 | C | TEN | TEN PS | CC011 | A-720-M/ULC | AV4 .1 | .3 | 107 | P 2 | 22 | | | | | | |
| 35 | 52 | C | TEN | TEN PS | CC012 | A-720-M/ULC | AV3 .0 | .3 | | P 2 | 21 | | | | | | |
| 39 | 54 | C | TEN | TEN PS | CC012 | A-720-M/ULC | AV3 .0 | .3 | | P 2 | 20 | | | | | | |
| 26 | 58 | C | TEN | TEN PS | CC015 | A-720-M/ULC | AV2 .0 | .5 | | P 2 | 26 | | | | | | |
| 30 | 61 | C | TEN | TEN PS | CC016 | A-720-M/ULC | AV2 .0 | .5 | | P 2 | 23 | | | | | | |
| 38 | 61 | C | TEN | TEN PS | CC011 | A-720-M/ULC | AV2 -.1 | .3 | 140 | P 2 | 22 | | | | | | |
| 21 | 62 | C | TEN | TEN PS | CC015 | A-720-M/ULC | AV2 .0 | .3 | | P 2 | 21 | | | | | | |
| 25 | 62 | C | TEN | TEN PS | CC015 | A-720-M/ULC | AV2 .0 | .3 | | P 2 | 20 | | | | | | |
| | | C | TEN | TEN PS | CC015 | A-720-M/ULC | AV3 .0 | .4 | | P 2 | 22 | | | | | | |
| 24 | 63 | C | TEN | TEN PS | CC016 | A-720-M/ULC | AV2 .0 | .3 | | P 2 | 20 | | | | | | |
| | | C | TEN | TEN PS | CC016 | A-720-M/ULC | AV3 .0 | .5 | | P 2 | 23 | | | | | | |
| 38 | 65 | C | TEN | TEN PC | CC013 | A-720-M/ULC | AV2 .0 | .3 | | P 2 | 22 | | | | | | |
| | | C | TEN | TEN PC | CC013 | A-720-M/ULC | AV3 .0 | .3 | | P 2 | 22 | | | | | | |

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



| | | | Extent | | | | 09/95 | | | | | | N/A | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Row | Col | Leg | Req | Tst/Note | Reel | Probe | Location | Volts | Deg | Ch | % | Diff | Location | Volts | Deg | Ch | % |
| | | C | TEH | TEH PC | CC013 | A-720-M/ULC | AV4 .0 | .5 | | P 2 | 25 | | | | | | |
| 38 | 71 | C | TEH | TEH PS | CC010 | A-720-M/ULC | AV3 .0 | .5 | | P 2 | 24 | | | | | | |
| 1 | 76 | C | 06C | 05C PS | CC020 | A-720-M/ULC | 02C 4.6 | 1.4 | 142 | 1 | 38 | | | | | | |
| | | C | 06C | 06C | CC038 | A-720-M/ULC | 02C 4.5 | 1.1 | 140 | 1 | 32 | | | | | | |

Number of RECORDS Selected from Current Outage : 47

Number of TUBES Selected from Current Outage : 30

ATTACHMENT 1

1. Request for Relief 6 seeks authorization to use Code Case N-532, "Alternative requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000." Justification for this relief is in support of using the Code Case, however, the submittal contains exceptions to the Code Case. Provide a technical discussion justifying those exceptions.

A. Code Case N-532 includes a footnote that says "All references to IWA-4000 and IWA-6000 used in this Case refer to the 1992 Edition." The approved Code Edition at Turkey Point is the 1989 Edition of Section XI. The Relief Request was worded so that Florida Power & Light (FPL) would be using a consistent approach to Section XI.

A comparison of the requirements between the 1989 and 1992 Section XI Codes shows differences to be minimal. FPL's existing procedures cover the necessary requirements.

Recently FPL received approval to use Code Case N-416-1. This Code Case requires that it be documented on NIS-2 forms when it is used. FPL will instead document this Code Case within the Repair/Replacement plan.

FPL will use the requirements of the 1992 Edition of Section XI. Compliance will not be a hardship.

During conversations with NRC personnel, a question was asked about the words "or similar" when addressing the OAR-1 and NIS-2 forms. FPL plans to use forms similar to those used in the Code Case. The square boxes surrounding text may be missing, and line breaks may be different. The text will be the same as the Code Case and placed in essentially the same position. Fill-in areas will be the same. Tables 1 through 3 will contain the same information in the same columns as shown in the Code Case. Differences will be due primarily to different word processors. The words "or similar" were added to allow this bit of leniency in the "look" of the form.

2. Requests for Relief 14 and 15 contain Class 1 and 2 piping examinations having limitations that prevent 100% ultrasonic examination. However it appears that extended beam paths were used on piping less than 12 inches in diameter in Unit 3. These extended beam paths were used from one side to obtain coverage in one direction. Provide a technical

position as to the possibility of using extended beam paths from the opposite side to obtain coverage in two directions on the subject welds in Units 3 and 4.

A. A review of Relief Request 14 shows there was an error in the table. The column that showed many of the 60 deg. 1-1/2 V coverage should have been marked in the 1/2 V column. This error was caused by reading the 1-1/2t coverage as 1-1/2V. Corrected tables are included.

The Turkey Point construction permit was issued before January 1, 1971. This was before Section XI requirements were mandated. 10CFR50.55a(g)(1) and 10CFR50.55a(g)(4) state the design and access requirements will be met to the extent practical, but are not required on units whose construction permit was issued before January 1, 1971. While Turkey Point has made an effort to gain access to many areas, it is not practical to redesign those areas with limited Section XI coverage.

The UT techniques for each weld were reviewed to determine if additional coverage could have been achieved. FPL's UT procedures require the examiner to consider whether additional coverage is necessary and can be achieved. Those alternate techniques were considered at the time of discovery. The alternate techniques considered were extending the calibration distance and using additional beam angles and/or modes. This has often provided the additional coverage needed to avoid the need for relief. Additional weld preparation by welding or metal removal is a modification of the examination area requiring significant engineering and construction personnel support. High radiation exposure and costs would be incurred in order to perform these types of modifications.

FPL has made reasonable efforts to meet Code requirements. The limitations were derived by graphically plotting the angles on the as-welded surfaces (when possible) and looking at actual and theoretical coverages that could be obtained by additional angles. In each case, the coverages obtained were considered the maximum practical. The alternative techniques would not have enhanced the coverage, nor added to the quality of the examination or safety of the system.

Copies of the original coverage sheets are attached with the revised Relief Requests.

3. **Request for Relief 17 requests to change two Examination Category C-B welds scheduled to be performed on Steam Generator "A" to Steam Generator "C". A schedule for Examination Categories C-A and C-B was attached to the request. These schedules differ from schedules and the philosophy approved in the Turkey Point Third 10-Year Interval Program Plan. Provide a discussion of Turkey Point selection criteria.**

A. FPL is requesting to change the examination schedule to reduce exposure and costs. This relief is requesting to substitute the examination of the "C" steam generator main steam nozzle weld in place of the same weld on the "A" steam generator. This weld has not been previously examined. The Turkey Point ISI Program did not address examining welds which had not previously been examined except when the originally scheduled weld was not accessible or if radiation exposure could be significantly reduced. This change would minimally reduce exposure, so the dose criteria in FPL Relief Request No. 9 was not being met. The purpose of requesting this change is to reduce the amount of time and work involved in removing insulation and to build scaffold on one steam generator instead of two. During the 1995 Unit 3 outage, it was discovered to be more efficient to examine the main steam nozzle to shell weld and the upper head to shell weld together rather than at different times. The amount of insulation removal was similar and the scaffold requirements were the same. Examining the two areas at the same time was cost effective and reduced some future radiation exposure.

This change in schedule would be grouping examinations together similar to what was requested in FPL's previously approved Relief Request No. 9. Scheduling the main steam nozzle to shell and inside radius examinations on the "C" generator would group them with weld 4-SGC-P, head to shell weld. This would allow all of the required examinations at the top of the steam generator to be performed at the same time (grouping of examinations). The bottom and middle sections of the Steam Generator are scheduled in this manner. This relief is requesting to perform the same type of efficient scheduling and examination for Unit 4 as was performed on Unit 3. By granting this relief, the schedule will be closer to compliance with Program B than requested in FPL Relief Request No. 9 and reduce costs and some radiation exposure.

ATTACHMENT 2
TO L-96-188

REVISED RELIEF REQUEST NO. 14

Relief Request No. 14, Limited Exams for 1994 Unit 3 Outage

A. Component Identification:

Class 1 and 2 pressure retaining similar and dissimilar metal welds in vessels and piping examined during the 1994 outage at Turkey Point Unit 3.

B. Examination Requirements:

Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, 1989 Edition with no Addenda

| Category | Item No. | Examination Requirements |
|---|---|---|
| B-D | B3.140 | Fig. IWB-2500-7(a through d), area defined by M-N-O-P |
| B-F | B5.70 | Fig. IWB-2500-8(b), weld and 1/2" to each side of the weld, 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |
| B-J | B9.11 | Fig. IWB-2500-8(b), weld and 1/2" to each side of the weld, 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |
| C-F-1 | C5.11 | Fig. IWC-2500-7(a), surface of weld and 1/2" of surface base metal (area A-B), 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |
| C-F-2 | C5.51 | Fig. IWC-2500-7(b), surface of weld and 1/2" of surface base metal (area A-B), 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |

ASME Code Case N-460 - Alternative Examination Coverage for Class 1 and Class 2 Welds

C. Relief Requested:

Relief is requested from the required code examination area during volumetric and surface examinations.

D. Basis for Relief:

Several welds examined during the 1994 outage did not achieve the required examination volume due to one or more factors:

1. Portions of the required volumetric area are inaccessible due to permanent physical obstructions.

2. Some welds could be examined from only one side due to the configuration of the component.

3. High attenuation of the ultrasonic sound.

The UT techniques for each weld were reviewed to determine if additional coverage could have been achieved. FPLs procedures require the examiner to consider whether additional coverage is necessary and practical. Those alternate techniques were investigated at the time of discovery. The alternate techniques considered were extending the calibration distance and using additional beam angles and/or modes. This has often provided the additional coverage needed to avoid relief. After considering the alternate UT techniques, it was determined they would have provided little or no additional coverage. The coverages obtained were the maximum practical by UT techniques.

If practical, physical obstructions were removed. In most cases, it was not possible to remove the obstruction without significant work, radiation exposure, and/or damage to the plant (i.e. pressurizer heaters at the surge line inside radius section.)

Additional weld preparation by welding or metal removal is a modification of the examination area requiring significant engineering and construction personnel support. High radiation exposure and costs would be incurred in order to perform these types of modifications. Radiography is impractical due to the amount of work being performed in the area on a 24 hour basis. This would result in numerous work related stoppages and increased exposure due to the shutdown and startup of other work in the area. Removal of water from the associated piping is not always possible, and when performed, increases the radiation dose rates in the area. It would be a significant hardship to perform weld or area modifications or radiography in order to increase examination coverage.

FPL has made reasonable efforts to meet Code requirements. Limitations were derived by graphically plotting the angles on replicas of the as welded surfaces (when possible) and looking at actual and theoretical coverages that could be obtained with additional angles. In each case, the coverages obtained were considered the maximum practical. The alternate techniques would not have enhanced the coverage, nor added to the quality of the examination or safety of the system.

FPL performed the examinations to the extent possible. Surface and volumetric examinations performed, along with the required system pressure tests, provide reasonable assurance of an acceptable level of quality and safety. System engineers perform walkdowns of every system on a periodic basis looking for leakage or other abnormal conditions. The attached table summarizes the percent of coverage achieved and references specific figures that show the extent of the limitations.

E. Alternative Examinations or Tests:

1. Volumetric and surface examinations were performed to the extent possible. Where practical, alternative examination techniques were performed.

2. System pressure tests as required by the Turkey Point Inservice Pressure Test Program were performed.

3. Monthly walkdowns by system engineers are performed on Class 2 systems to check for leakage, piping configuration, and/or damage. During outages, system engineers walkdown Class 1 and Class 2 systems inside containment. This walkdown is performed to look for system anomalies which could effect plant performance.

The examination volume achieved by surface and/or ultrasonic examination, combined with the system pressure tests and system engineer walkdowns, provide an acceptable level of quality and safety. If permanent obstructions are removed, FPL will examine those areas to the extent practical.

F. Implementation Schedule:

These examinations were performed during the first outage of the first period of the third inservice inspection interval, from April 4, 1994 through May 4, 1994.

G. Attachments

Table showing areas where limited examinations were performed and the extent of coverage.

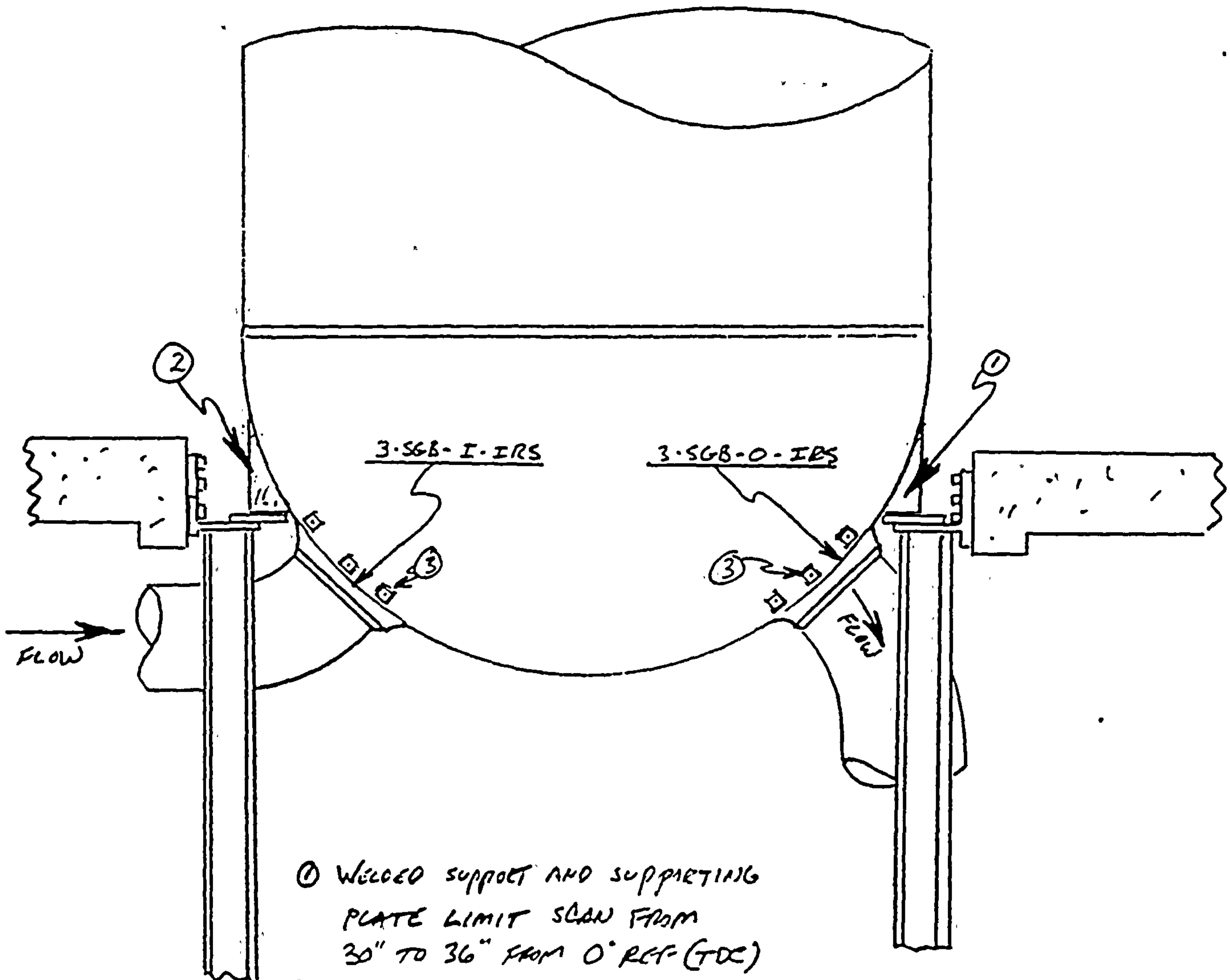
Sketches of areas with limited examinations.

| Catgy/ Item No. | Component ID | NDE Technique | Angle and Technique | | | Configuration and/or Limitations | Fig. No. | Comments /% Coverage |
|---|---|---|---|---|---|---|---|---|
| | | | ½ V | Full V | 1½ V | | | |
| B-D B3.140 | 3-SGB-I-IRS<br>3-SGB-O-IRS | UT | 45 | - | - | Welded on pads, support, and IRS configuration limited the examination | 94-3-1 | 70% coverage achieved in two circumferential directions |
| B-D B3.120 | 3-SRGN-01-IR | UT | 30<br>60 | -<br>- | -<br>- | heater penetrations limited the examination | 94-3-2 | 55% coverage from 2 directions |
| B-F B5.70 | 31"-RCS-1302-5 | UT | 45 | - | - | Steam generator nozzle to elbow, configuration limits the examination, elbow material is highly attenuative | 94-3-3 | 75% from the elbow side, 0% from the nozzle side, only 1/2 V examination possible from the elbow side |
| B-J B9.11 | 31"-RCS-1302-10 | UT | 45 | - | - | Elbow to pump casing, configuration limits the examination, elbow and pump material are highly attenuative | 94-3-4 | 75% from the elbow side, 0% from the pump side, only 1/2 V examination possible from the elbow side |
| B-J B9.11 | 29"-RCS-1305-3 | UT | 45 | - | - | elbow to pipe, Branch connection limits 18" of examination, configuration limits the examination, elbow material is highly attenuative | 94-3-5 | 93% from pipe side, 69% from the elbow side |
| B-F B5.70 | 29"-RCS-1305-4 | UT | 45 | - | - | Nozzle to elbow, configuration limits the examination, elbow material is highly attenuative | 94-3-6 | 43% from elbow side, 2% from the nozzle side |
| B-J B9.11 | 27.5"-RCS-1306-11 | UT | 45 | - | - | Pump casing to pipe, configuration limits the examination, pump material is highly attenuative | 94-3-7 | 0% from the pump side, 100% from pipe side, only 1/2 V examination possible from the pipe side |

| Catgy/ Item No. | Component ID | NDE Technique | Angle and Technique ½ V | Full V | 1½ V | Configuration and/or Limitations | Fig. No. | Comments /% Coverage |
|---|---|---|---|---|---|---|---|---|
| B-J B9.11 | 12"-RC-1301-1 | UT | 45 60 | - - | - - | Branch connection to pipe, configuration limits the examination | 94-3-8 | 100% from the pipe side, 67% from the branch connection side |
| B-J B9.11 | 10"-SI-1302-1 | UT | 45 60 | - - | - - | Elbow to branch connection, configuration limits the examination | 94-3-9 | 100% from the elbow side, 85% from the valve side |
| B-J B9.11 | 10"-SI-1302-4 | UT | 45 60 | - - | - - | Elbow to valve 3-875B, configuration and branch connection limits the examination | 94-3-10 | 85% from the elbow side, 25% from the valve side |
| C-F-1 C5.11 | 8"-SI-2309-22 | UT | 45 60 | - - | - - | Pipe to tee, configuration limits the examination | 94-3-11 | 100% from the pipe side, 47% from the tee side |
| C-F-1 C5.11 | 8"-SI-2309-24 | UT | 45 60 | - - | - - | Pipe to valve 3-876E, configuration limits examination | 94-3-12 | 100% from the pipe side, 0% from the valve side |
| C-F-2 C5.51 | 6"-BDA-2301-8 | UT | - | 60 | 45 | Tee/wye to pipe, configuration and rough crown limits examination | 94-3-13 | 100% from pipe side, 50% from tee/wye side |

PTN 3

ZONE 004 / SKETCH 3-V09B

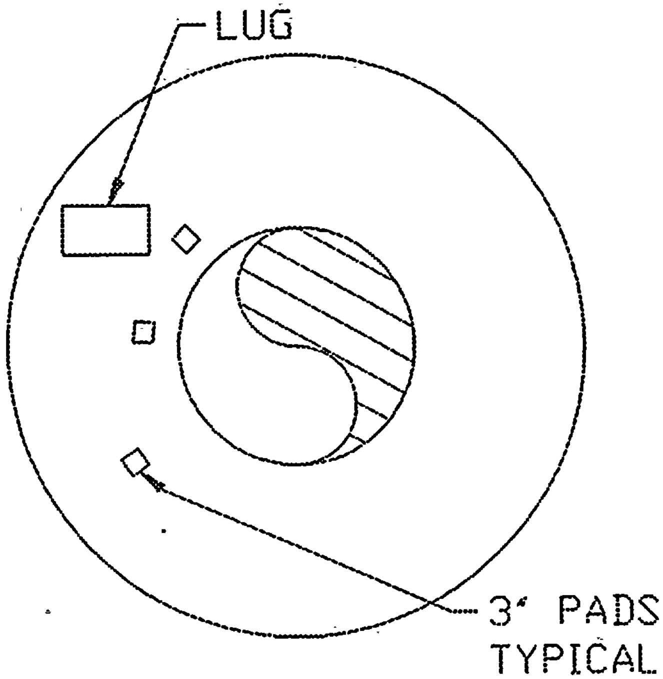



① WELDED SUPPORT AND SUPPORTING
PLATE LIMIT SCAN FROM
30" TO 36" FROM 0° REF (TDC)

② WELDED SUPPORT AND SUPPORTING
PLATE LIMIT SCAN FROM
35" TO 40" CW FROM 0 REF (TDC)

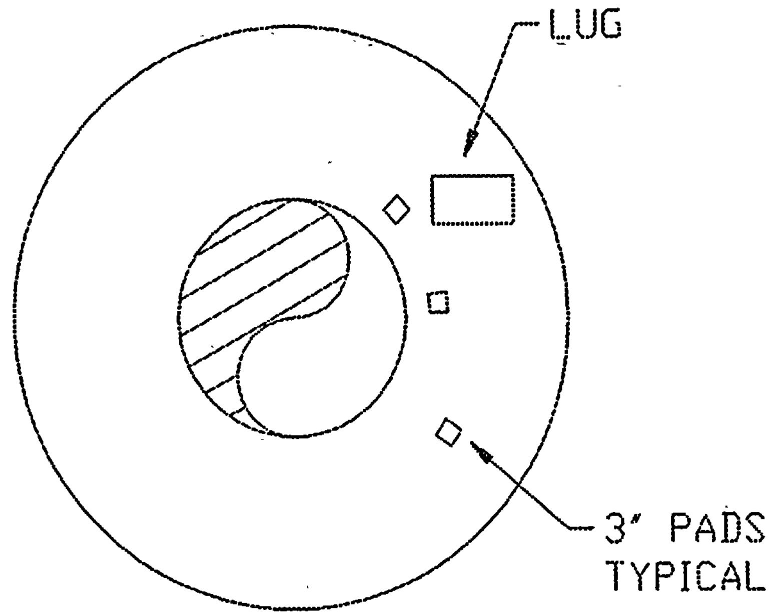
③ TYPICAL 2½ x 2½ WELDED PADS LIMIT SCANS
AS FOLLOWS (IN INCHES CW FROM 0° REF)
<u>HOT LEG</u> - 30", 47", 63", 84" + 107".
<u>INTERMEDIATE LEG</u> - 58", 44", 80", + 135".

94-3-1 Sheet 1 of 2




3-SGB-I-IRS

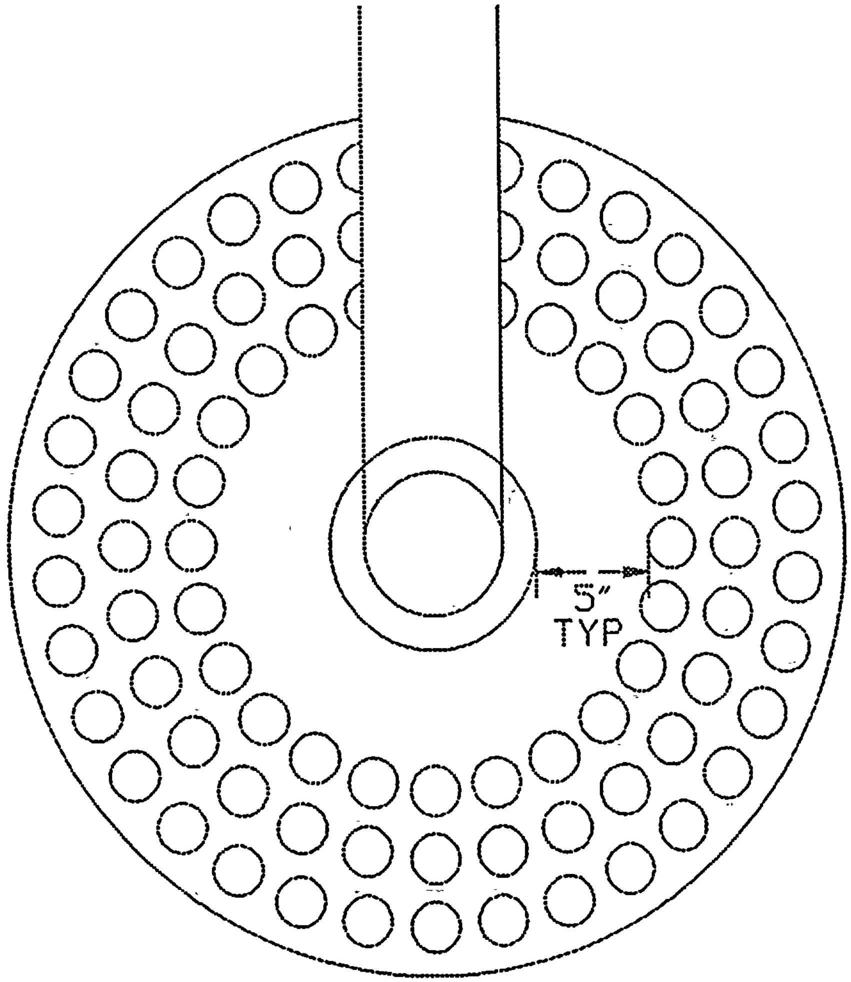
INLET NOZZLE INNER RADIUS




3-SGB-O-IRS

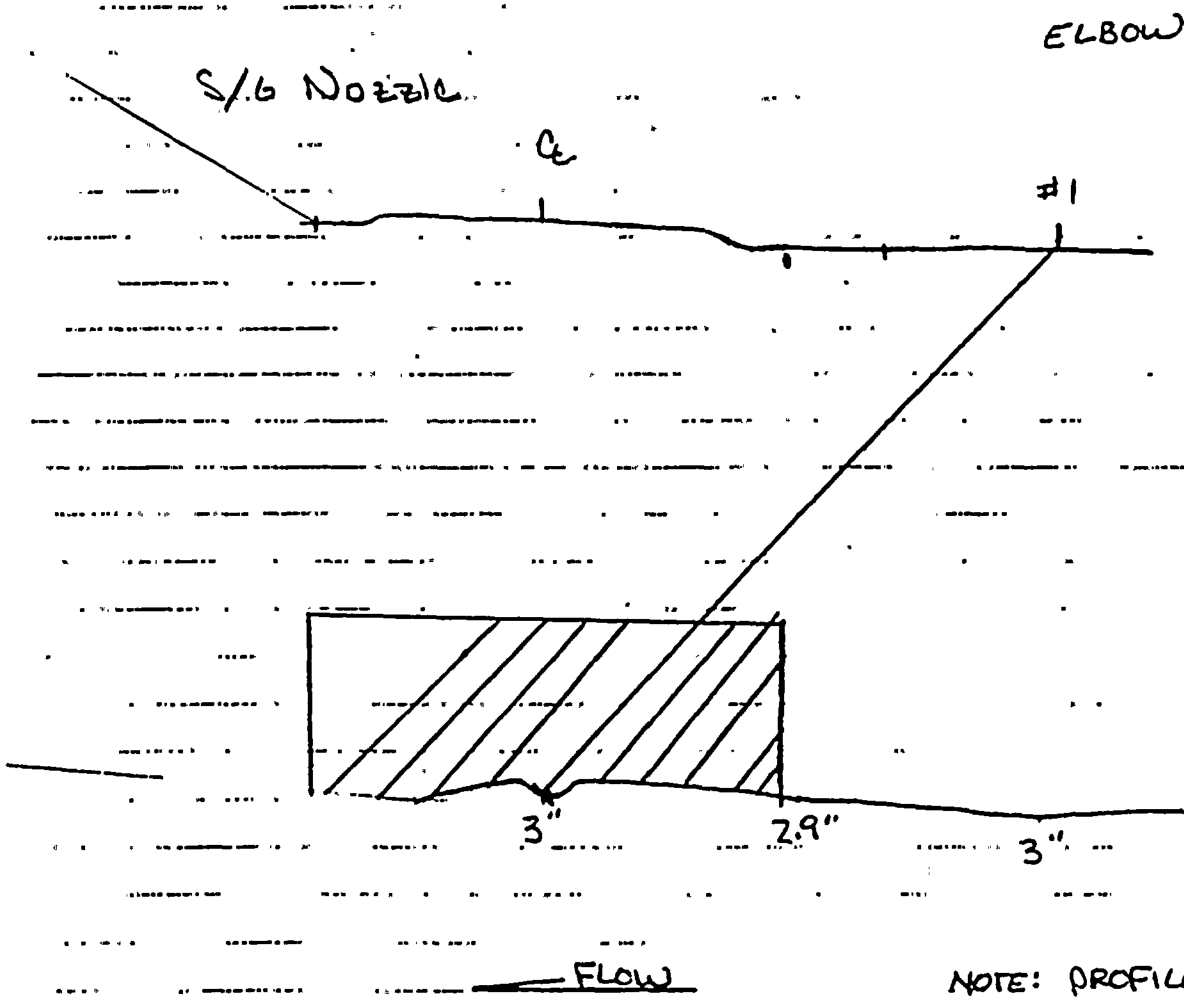
OUTLET NOZZLE INNER RADIUS

94-3-1 Sheet 2 of 2




PRESSURIZER SURGE NOZZLE
3-SRGN-01-IR

31-RCS-1302-5

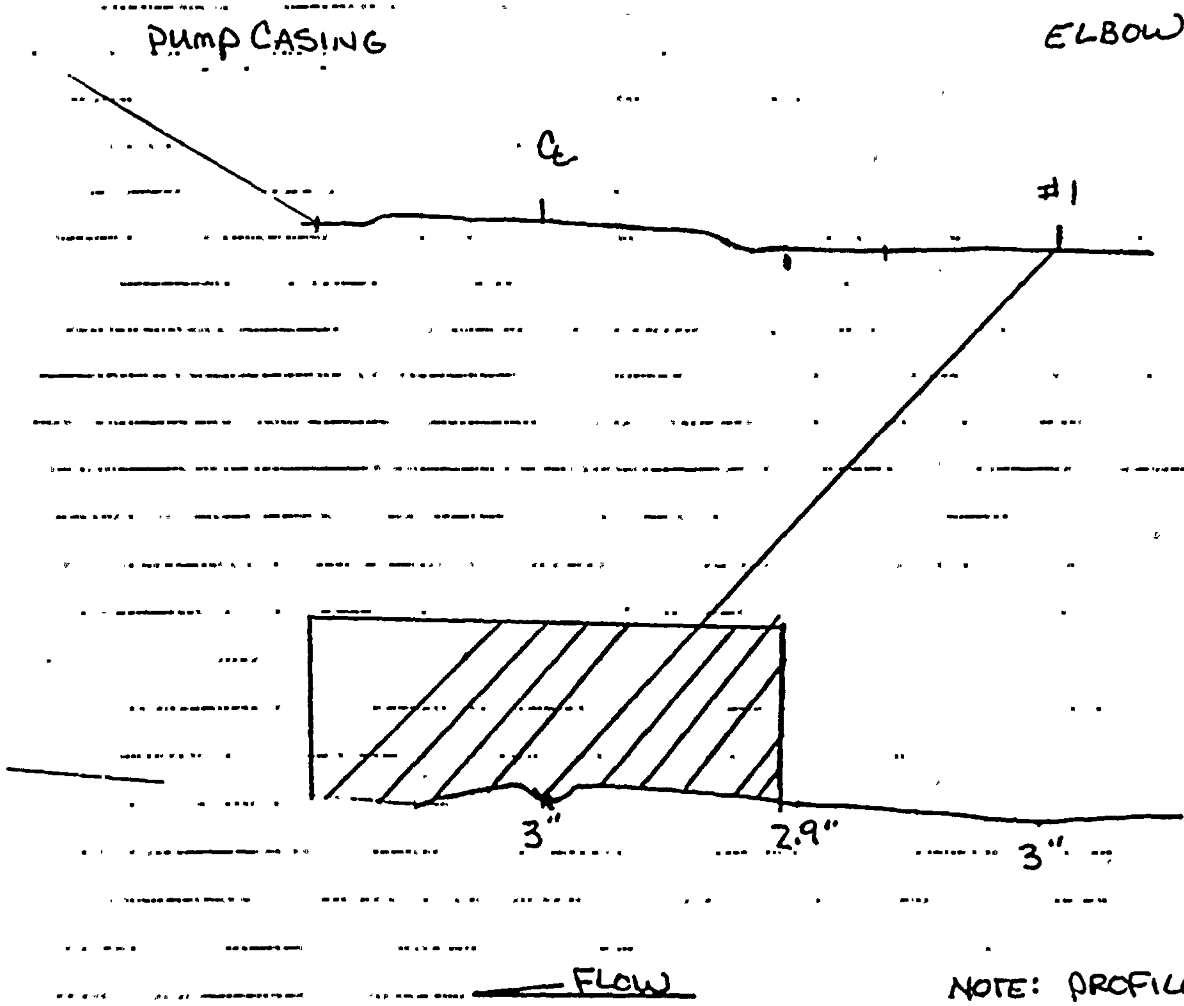



NOTE: PROFILE VERIFIED WITH 0°.

75% from elbow/pipe side
0% from nozzle side

94-3-3

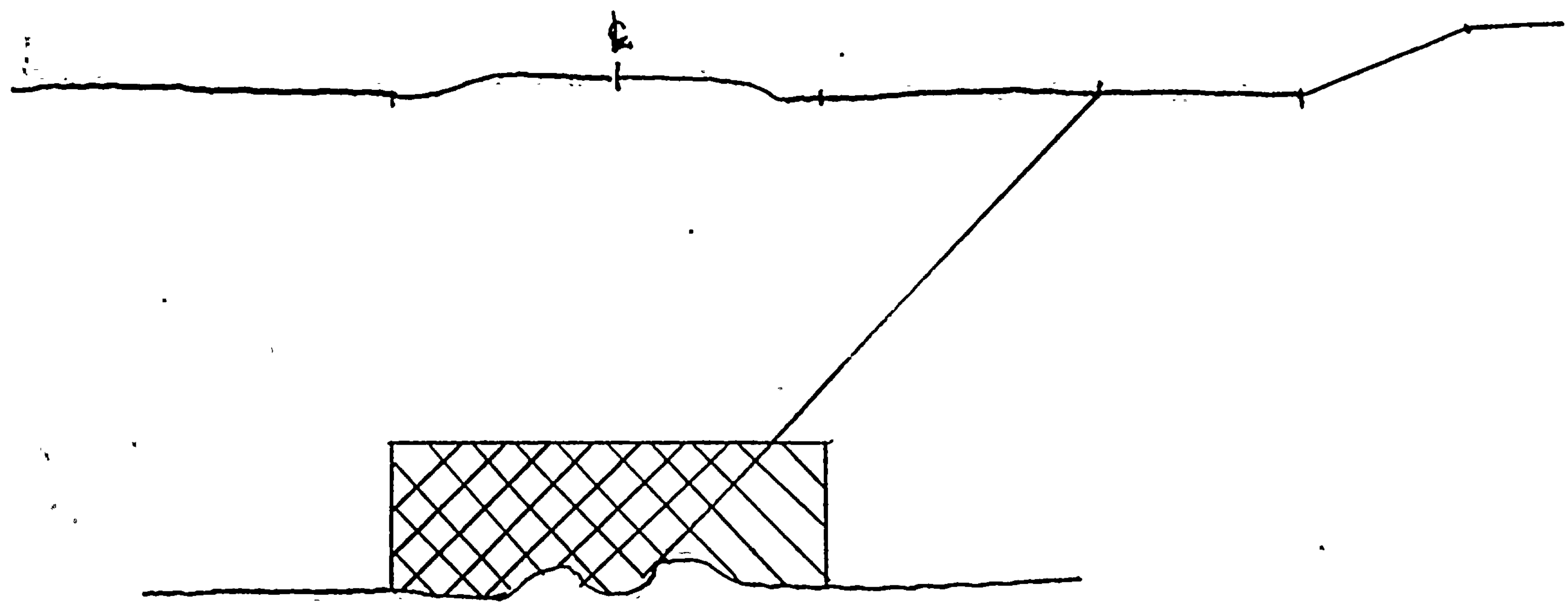
31"-RCS-1302-1D




NOTE: PROFILE VERIFIED WITH 0°. ~~AB~~ 3-14-91

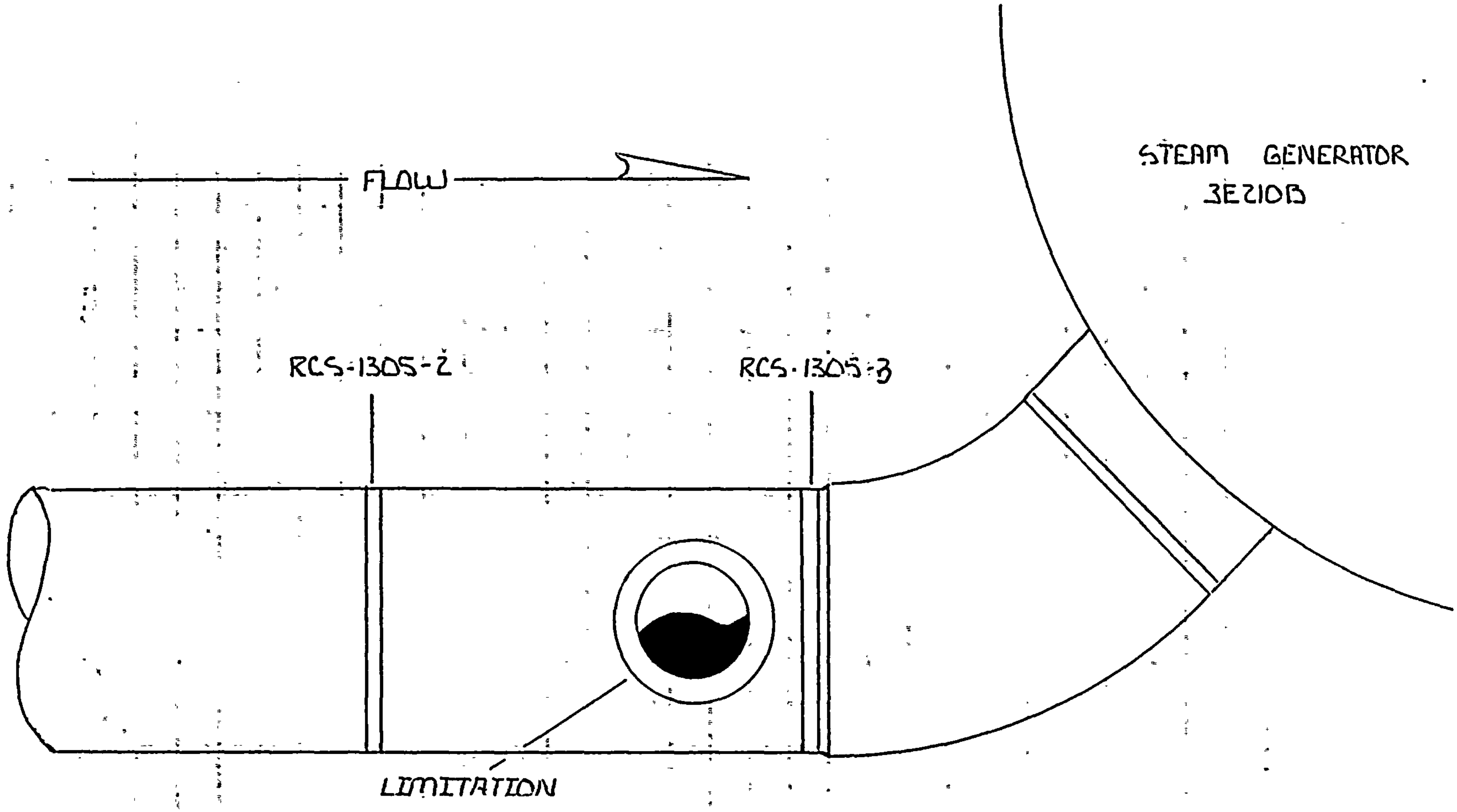
75% from elbow/pipe side
0% from nozzle side

29"- RCS-130S.3



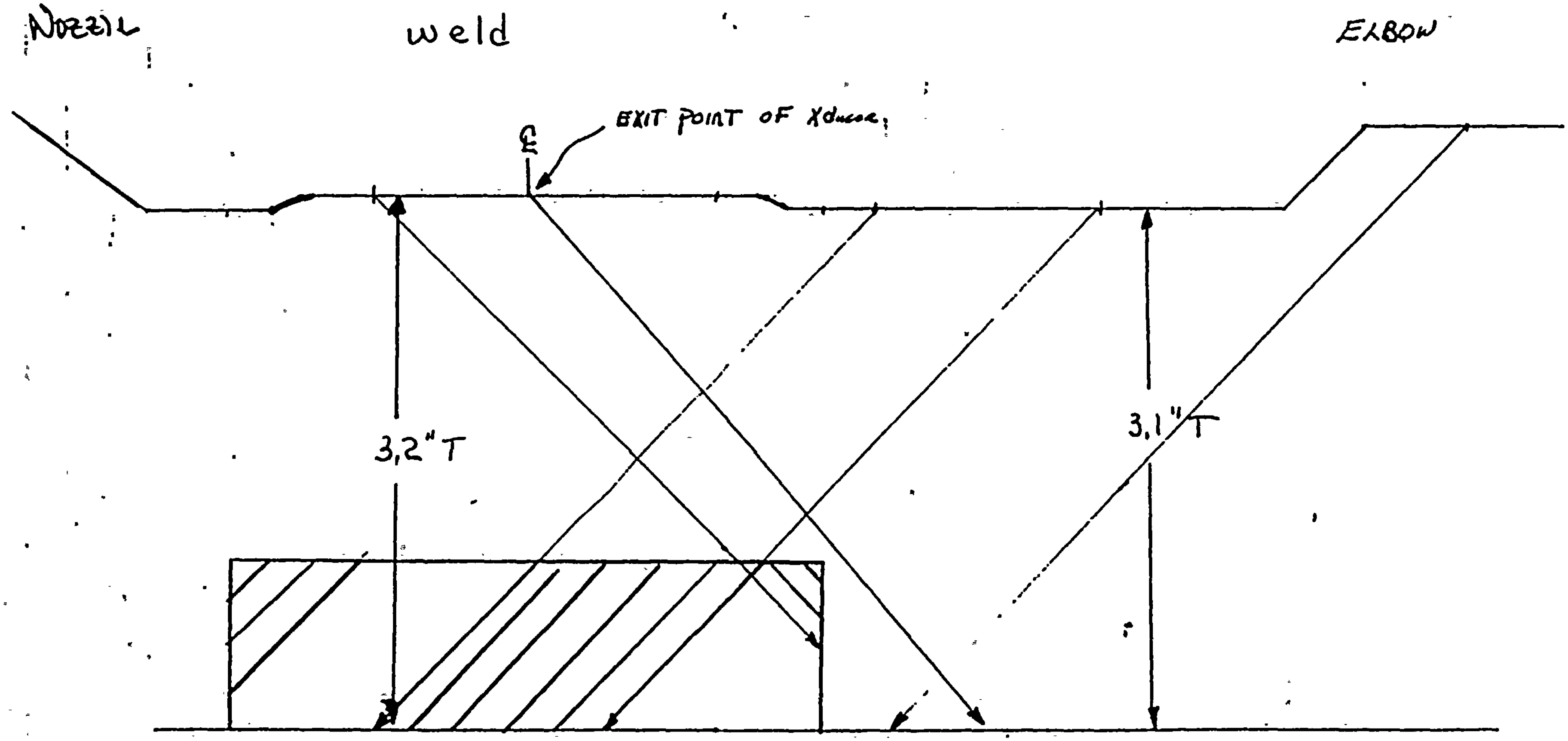
93% from Pipe Side (Branch Conn Limitation)
69% from Elbow Side (OD Config)

13




LIMITATION - 12" BRANCH CONNECTION RC-1301
2" FROM TOE OF WEID AT 18" TO
36" CW.

94-3-5 Sheet 2 of 2

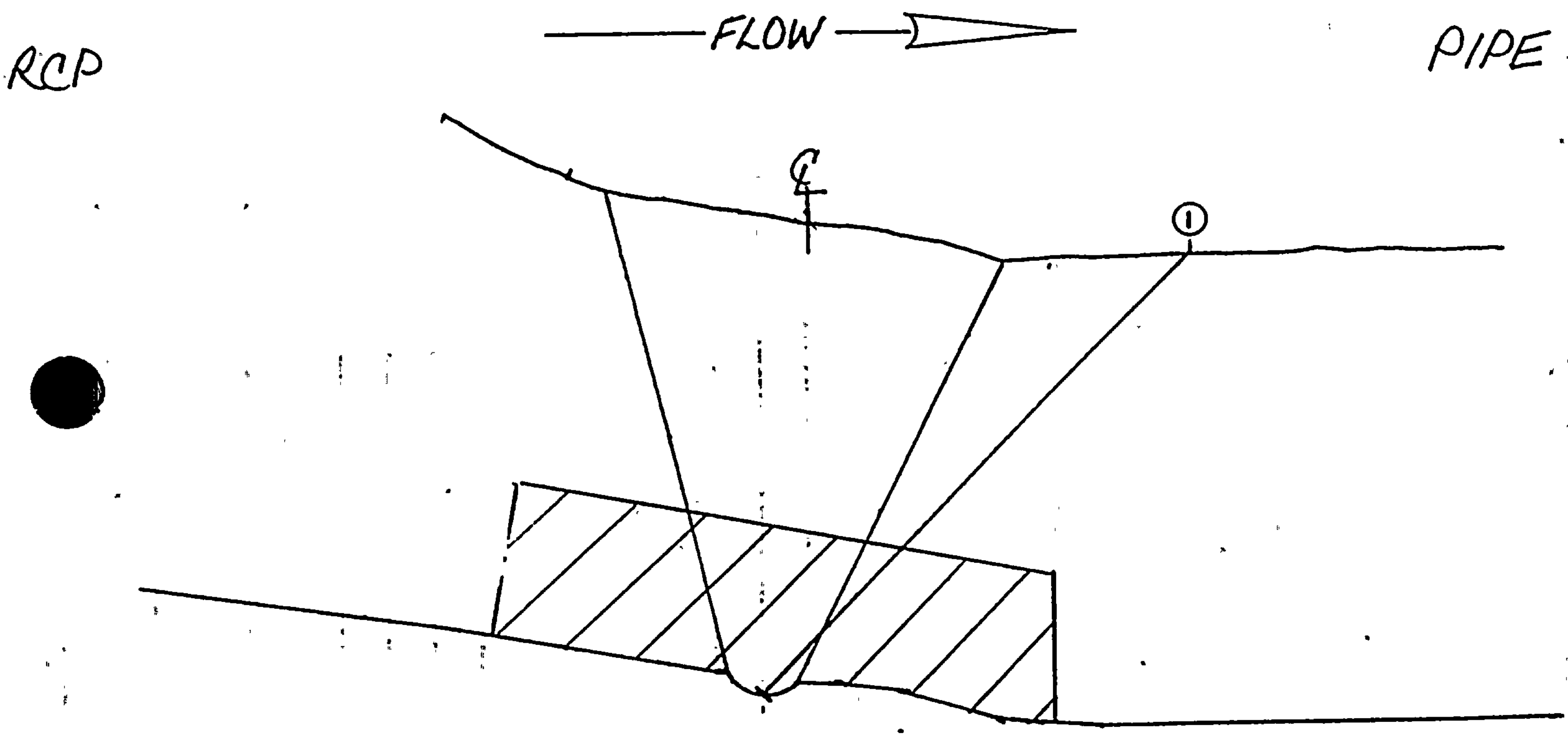



29"-RCS-1305-4

43% CRV from Elbow side
2% CRV from Nozzle Side

94-3-6

27.5"-RCS-1306-11



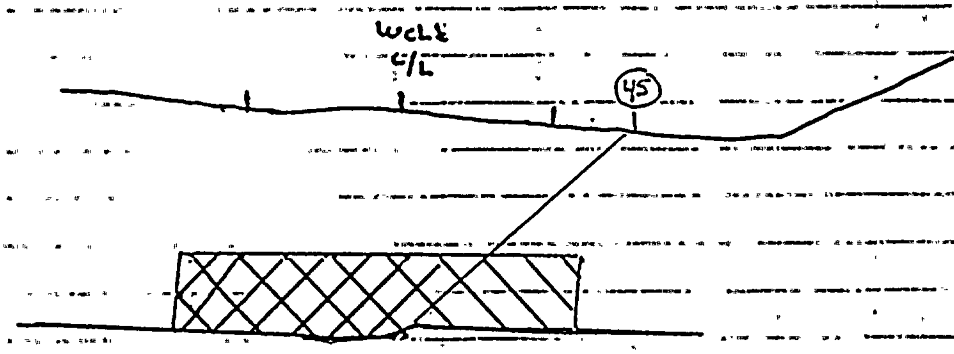

100% from Pipe Side
0% from Pump Side

94-3-7

CRV Sketch For Weld
12"-RC-1301-1

Pipe Side

NOZZLE
REDUCER Side



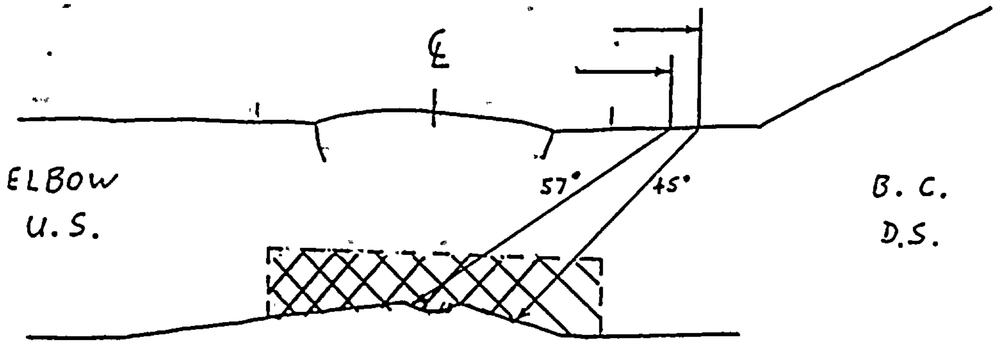

100% from Pipe Side
64% from nozzle Side
[signature] UT-III

94-3-8

WELD No. 10"-SI-1302-1

CRV SKETCH

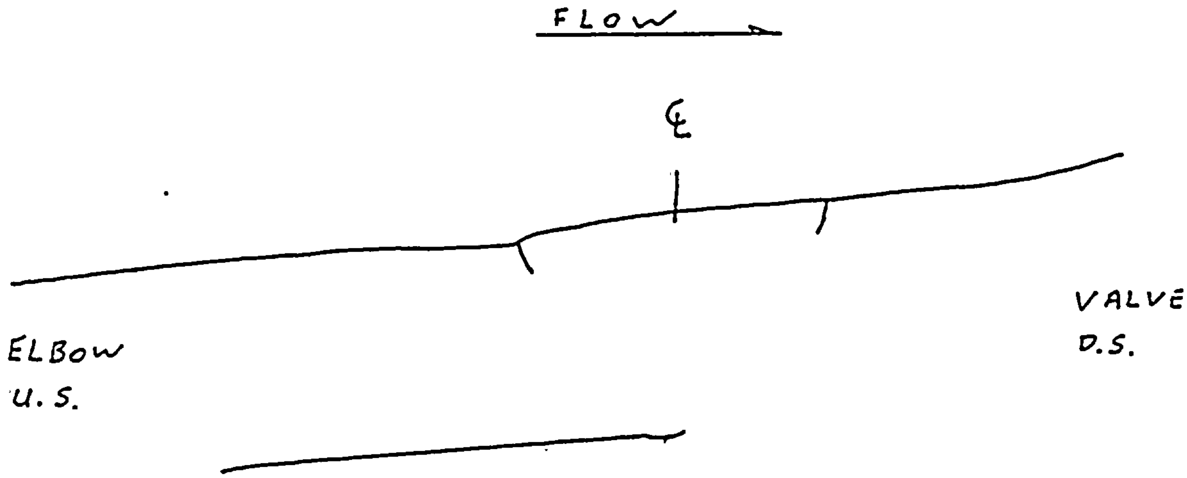
FLOW ▷




CRV
100% from elbow side
85% from Branch Side

94-3-9

10" - SI - 1302 - 4 CRV SKETCH




- AXIAL EXAMINATIONS FROM U.S. SIDE OBSTRUCTED FROM L = 6" CW TO L = 10.5" CW DUE TO BRANCH CONNECTION, AND RESTRICTED TO WELD CROWN ONLY FROM D.S. SIDE DUE TO VALVE GEOMETRY

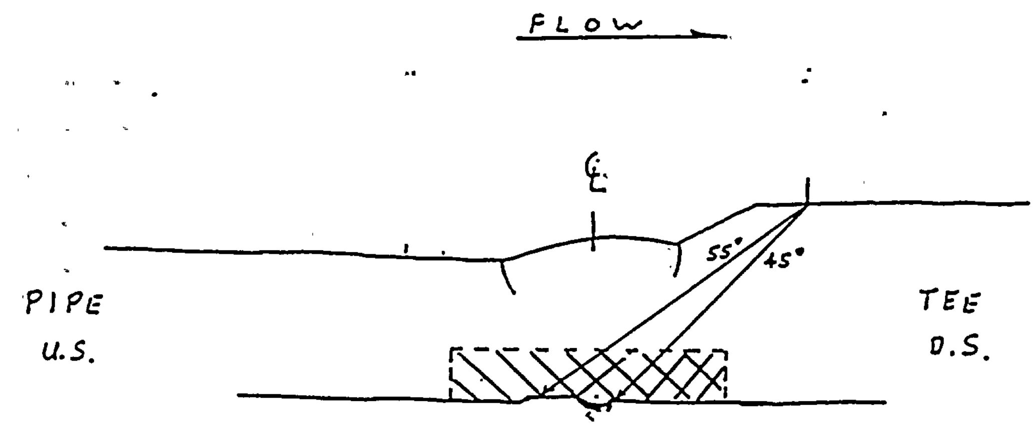
- CIRC EXAMINATION ON U.S. SIDE OBSTRUCTED FROM L = 6" CW TO L = 10.5" CW DUE TO BRANCH CONNECTION, WAS PERFORMED ON REMAINDER OF U.S. SIDE AND WELD CROWN ONLY, IN TWO DIRECTIONS.

- $L_{TOTAL}$ = 34"

85% from Elbow side
25% from Valve side

94-3-10

8"-SI-2309-22 CRV SKETCH

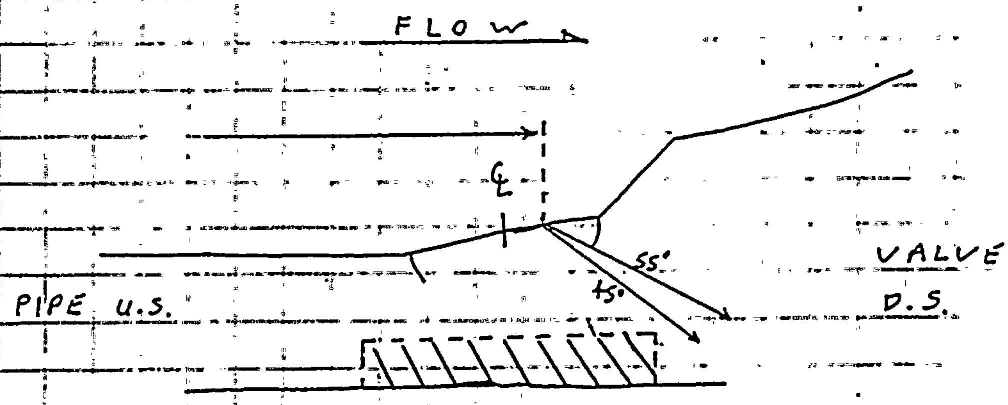



- AXIAL EXAM FROM D.S. SIDE PARTIALLY OBSTRUCTED BY BEVEL ON TEE ADJACENT TO WELD TOE.
- AXIAL EXAM FROM D.S. SIDE OBSTRUCTED BY TEE BRANCH FROM L= 11.5" cw TO L= 15.5" cw
- L TOTAL = 27" 85% of circ examined from TEE Side

100% CRV from Pipe Side
47% CRV from TEE Side

94-3-11

8"-SI-2309-24 CRV SKETCH




• AXIAL EXAM PERFORMED FROM U.S. SIDE ONLY, DUE TO VALVE CONFIGURATION.

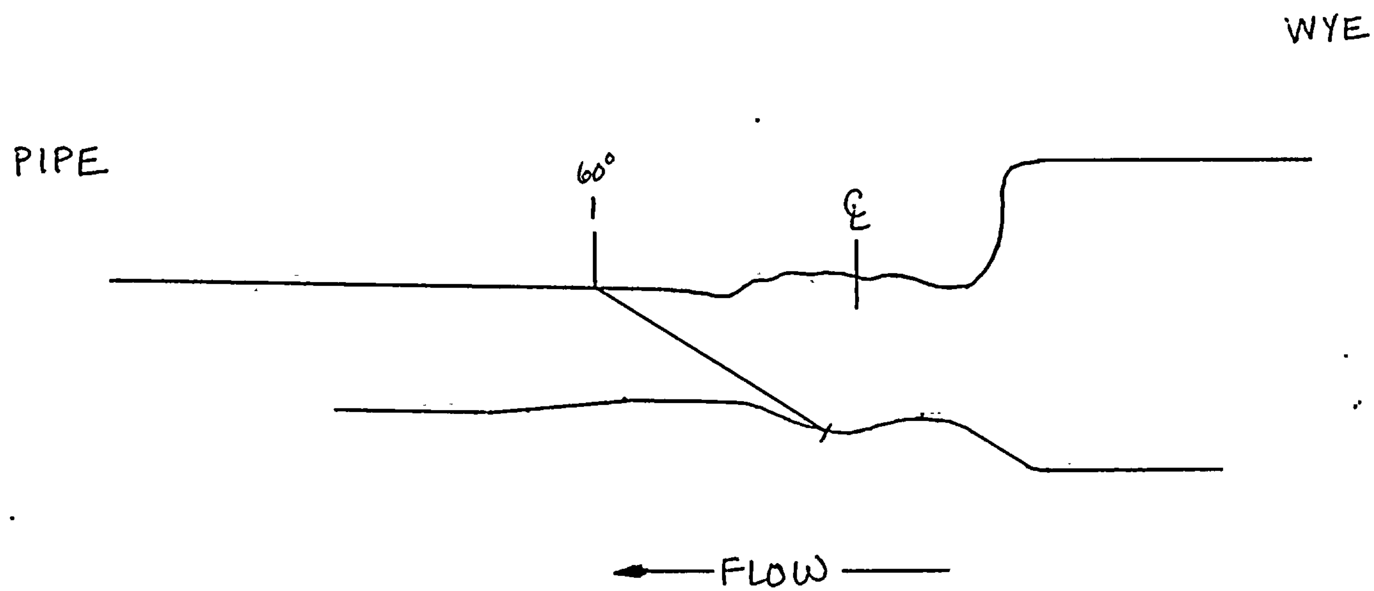
• CIRCUMFERENCIAL EXAM PERFORMED IN TWO DIRECTIONS ON U.S. SIDE AND WELD CROWN ONLY, DUE TO VALVE CONFIGURATION.

• 55° ANGLE MEASURED IN COMPONENT.

100% from Pipe Side
0% from Valve Side

94-3-12

EAM GENERATOR A
BLOWDOWN
ZONE: 103
6"BDA 2301-8




PIPE TO WYE

— ONE-SIDED EXAM DUE TO PIPE TO WYE CONFIGURATION

Note: Rough Weld crown and config Limitation.
100% from Pipe side
50% from wye side

ATTACHMENT 3
TO L-96-188

REVISED RELIEF REQUEST NO. 15

Relief Request No. 15, Limited Exams for 1994 Unit 4 Outage

A. Component Identification:

Class 1 pressure retaining similar and dissimilar metal welds in vessels and piping found during the 1994 outage at Turkey Point Unit 4.

B. Examination Requirements:

Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, 1989 Edition with no Addenda

| Category | Item No. | Examination Requirements |
|---|---|---|
| B-D | B3.140 | Fig. IWB-2500-7(a through d), area defined by M-N-O-P |
| B-F | B5.70 | Fig. IWB-2500-8(b), weld and 1/2" to each side of the weld |
| B-J | B9.11 | Fig. IWB-2500-8(b), weld and 1/2" to each side of the weld, 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |
| | B9.31 | Fig. IWB-2500-10, weld and 1/2t or 1" (whichever is less) to each side of the weld, 1/3t from the inside surface out to 1/4" from a line drawn from the toe of the outside surface weld crown (area C-D-E-F) |

ASME Code Case N-460 - Alternative Examination Coverage for Class 1 and Class 2 Welds

C. Relief Requested:

Relief is requested from the required code examination area during volumetric and surface examinations.

D. Basis for Relief:

Several welds examined during the 1994 outage did not achieve the required examination volume due to one or more factors:

1. Portions of the required volumetric area are inaccessible due to permanent physical obstructions.

2. Some welds could be examined from only one side due to the configuration of the component.

3. High attenuation of the ultrasonic sound.

The UT techniques for each weld were reviewed to determine if additional coverage could have been achieved. FPL's procedures require the examiner to consider whether additional coverage is necessary and practical. Those alternate techniques were investigated at the time of discovery. The alternate techniques considered were extending the calibration distance and using additional beam angles and/or modes. This has often provided the additional coverage needed to avoid relief. After considering the alternate UT techniques, it was determined they would have provided little or no additional coverage. The coverages obtained were the maximum practical by UT techniques.

If practical, physical obstructions were removed. In most cases, it was not possible to remove the obstruction without significant work, radiation exposure, and/or damage to the plant (i.e. pressurizer heaters at the surge line inside radius section.)

Additional weld preparation by welding or metal removal is a modification of the examination area requiring significant engineering and construction personnel support. High radiation exposure and costs would be incurred in order to perform these types of modifications. Radiography is impractical due to the amount of work being performed in the area on a 24 hour basis. This would result in numerous work related stoppages and increased exposure due to the shutdown and startup of other work in the area. Removal of water from the associated piping is not always possible, and when performed, increases the radiation dose rates in the area. It would be a significant hardship to perform weld or area modifications or radiography in order to increase examination coverage.

FPL has made reasonable efforts to meet Code requirements. Limitations were derived by graphically plotting the angles on replicas of the as welded surfaces (when possible) and looking at actual and theoretical coverages that could be obtained with additional angles. In each case, the coverages obtained were considered the maximum practical. The alternate techniques would not have enhanced the coverage, nor added to the quality of the examination or safety of the system.

FPL performed the examinations to the extent possible. Surface and volumetric examinations performed, along with the required system pressure tests, provide reasonable assurance of an acceptable level of quality and safety. System engineers perform walkdowns of every system on a periodic basis looking for leakage or other abnormal conditions. The attached table summarizes the percent of coverage achieved and references specific figures that show the extent of the limitations.

E. Alternative Examinations or Tests:

1. Volumetric and surface examinations were performed to the extent possible. Where practical, alternative examination techniques were performed.

2. System pressure tests as required by the Turkey Point Inservice Pressure Test Program were performed.

3. During outages, system engineers walkdown Class 1 and Class 2 systems inside containment. This walkdown is performed to look for system anomalies which could effect plant performance.

The examination volume achieved by surface and/or ultrasonic examination, combined with the system pressure tests and system engineer walkdowns, provide an acceptable level of quality and safety. If permanent obstructions are removed for other reasons, FPL will examine those areas that become accessible to the extent practical.

F. Implementation Schedule:

These examinations were performed during the first outage of the first period of the third inservice inspection interval, from October 3, 1994 through November 14, 1994.

G. Attachments

Table showing areas where limited examinations were performed and the extent of coverage.

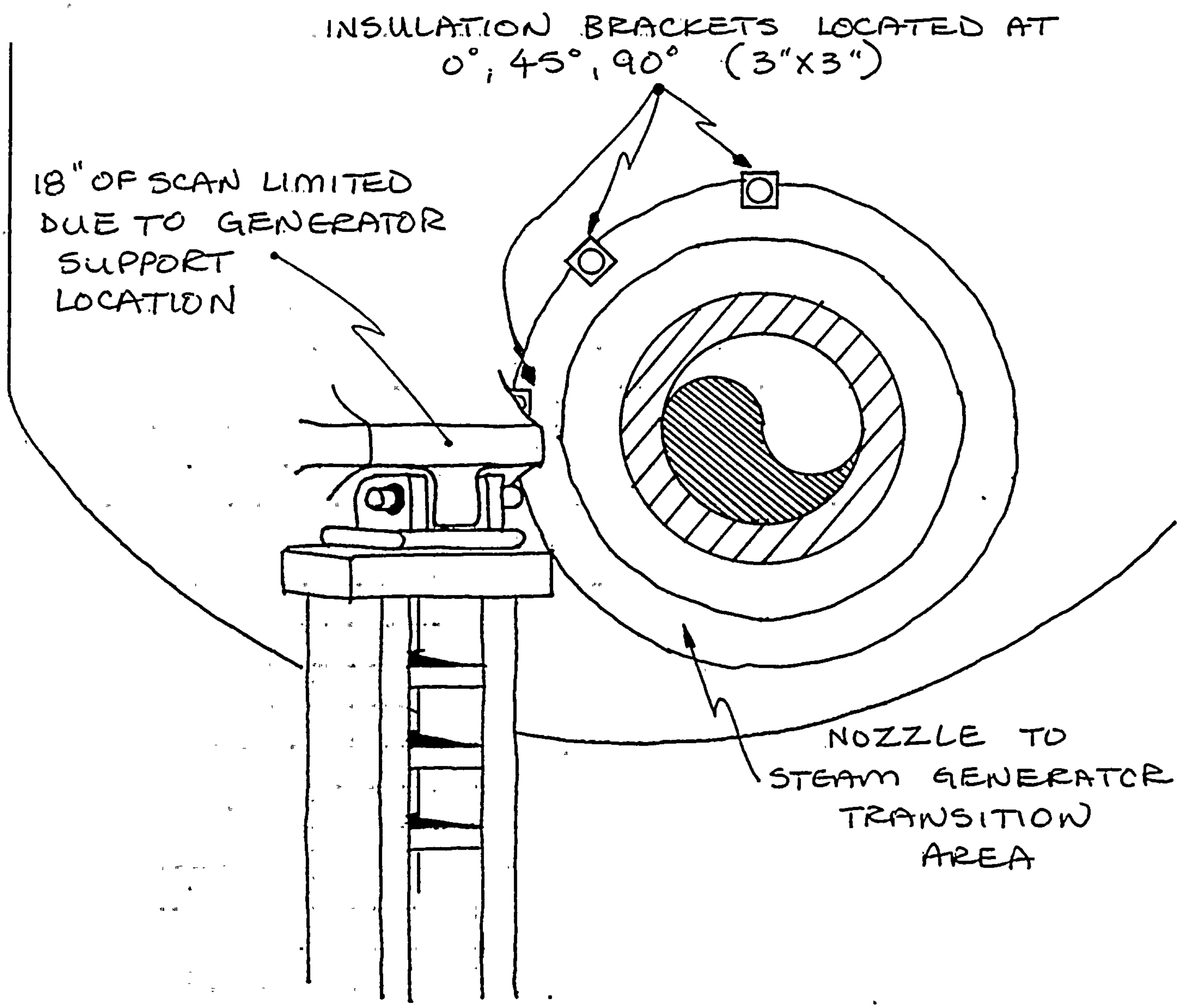
Sketches of areas with limited examinations.

| Catgy/ Item No. | Component ID | NDE Technique | Angle and Technique ½ V | Full V | 1½ V | Configuration and/or Limitations | Fig. No. | Comments / % Coverage |
|---|---|---|---|---|---|---|---|---|
| B-D B3.140 | 4-SGA-I-IRS 4-SGA-O-IRS | UT | 45 | - | - | Inner radius section, insulation brackets, support, and nozzle transition area limit the examination area | 94-4-1 | 75% coverage achieved in two circumferential directions |
| B-F B5.70 | 31"-RCS-1401-5 | UT | 45 | - | - | Steam generator nozzle to elbow, elbow material is highly attenuative, configuration limits examination | 94-4-2 | 0% from the nozzle side, 62% from the elbow side, only 1/2 V examination possible from the elbow side |
| B-J B9.11 | 31"-RCS-1401-8 | UT | 45 | - | - | Elbow to pipe, configuration limits the examination, elbow material is highly attenuative | 94-4-3 | 64% from the pipe side, 88% from the elbow side, only 1/2 V examination possible from both sides |
| B-J B9.11 | 31"-RCS-1401-10 | UT | 45 | - | - | Elbow to reactor coolant pump casing, configuration limits examination, elbow and pump material are highly attenuative | 94-4-4 | 57% from the elbow side, 0% from the pump side, only 1/2 V examination possible from the elbow side |
| B-F B5.70 | 29"-RCS-1404-4 | UT | 45 | - | - | Elbow to steam generator nozzle, configuration limits examination area, elbow material is highly attenuative | 94-4-5 | 74% from the elbow side, 0% from the nozzle side, only 1/2 V examination possible from the elbow side |
| B-J B9.31 | 29"-RCS-1404-18 | UT | 45 70 | - | - | 14" branch connection, configuration limits coverage, pipe material is highly attenuative | 94-4-6 | 0% from the pipe side, 100% from the branch connection side, only 1/2 V examination possible from the pipe side |

| Catgy/ Item No. | Component ID | NDE Technique | Angle and Technique ½ V | Full V | 1½ V | Configuration and/or Limitations | Fig. No. | Comments / % Coverage |
|---|---|---|---|---|---|---|---|---|
| B-J B9.11 | 27.5"-RCS-1407-11 | UT | 45 | - | - | Reactor coolant pump case to pipe, configuration limits the examination , pump material is highly attenuative | 94-4-7 | 18% from the pump side, 41% from the pipe side, only 1/2 V examination possible from the pipe side |
| B-J B9.31 | 27.5"-RCS-1407-20 | UT | 45 70 | - | - | 10" branch connection, configuration limits the examination , pipe material is highly attenuative | 94-4-8 | 100% from the nozzle side, 0% from the main loop pipe side, only 1/2 V examination possible from the pipe side |
| B-J B9.31 | 29"-RCS-1405-21 | UT | 45 70 | - | - | 12" branch connection, configuration limits the examination, pipe material is highly attenuative | 94-4-9 | 75% branch connection side, 0% from the pipe side, only 1/2 V examination possible from the pipe side |
| B-J B9.31 | 27.5"-RCS-1406-18 | UT | 45 70 | - | - | 10" branch connection, configuration limits the examination, pipe material is highly attenuative | 94-4-10 | 100% from the nozzle side, 0% from the main loop pipe side, only 1/2 V examination possible from the pipe side |
| B-J B9.31 | 27.5"-RCS-1409-16 | UT | 45 60 | - | - | 4" branch connection, configuration limits the examination, pipe material is highly attenuative | 94-4-11 | 100% from the nozzle side, 0% from the main loop pipe side, only 1/2 V examination possible from the pipe side |
| B-J B9.31 | 27.5"-RCS-1409-17 | UT | 45 70 | - | - | 10" branch connection, configuration limits the examination, pipe material is highly attenuative | 94-4-12 | 100% from the nozzle side, 0% from the main loop pipe side, only 1/2 V examination possible from the pipe side |

| Catgy/ Item No. | Component ID | NDE Technique | Angle and Technique ½ V | Full V | 1½ V | Configuration and/or Limitations | Fig. No. | Comments / % Coverage |
|---|---|---|---|---|---|---|---|---|
| B-J B9.11 | 14"-RHR-1401-1 | UT | 45 60 | - | - | Branch connection to elbow, configuration limits the examination | 94-4-13 | 70% from the elbow side, 35% from the branch connection side |
| B-J B9.11 | 14"-RHR-1401-5 | UT | 45 60 | - | - | Pipe to valve, configuration limits the examination | 94-4-14 | 100% from the pipe side, 0% from the valve side |
| B-J B9.11 | 14"-RHR-1401-6 | UT | 45 60 | - | - | Valve to pipe, configuration limits the examination | 94-4-15 | 100% from the pipe side, 0% from the valve side |
| B-J B9.11 | 14"-RHR-1401-9 | UT | 45 60 | - | - | Elbow to pipe, configuration limits the examination | 94-4-16 | 53% from the pipe side, 54% from the elbow side |
| B-J B9.11 | 10"-SI-1401-14 | UT | 45 60 | - | - | Valve to pipe, configuration limits the examination | 94-4-17 | 100% from the pipe side, 26% from the valve side |
| B-J B9.11 | 10"-SI-1401-18 | UT | 45 60 | - | - | Pipe to branch connection, configuration limits the examination | 94-4-18 | 100% from the pipe side, 18% from the branch connection side |

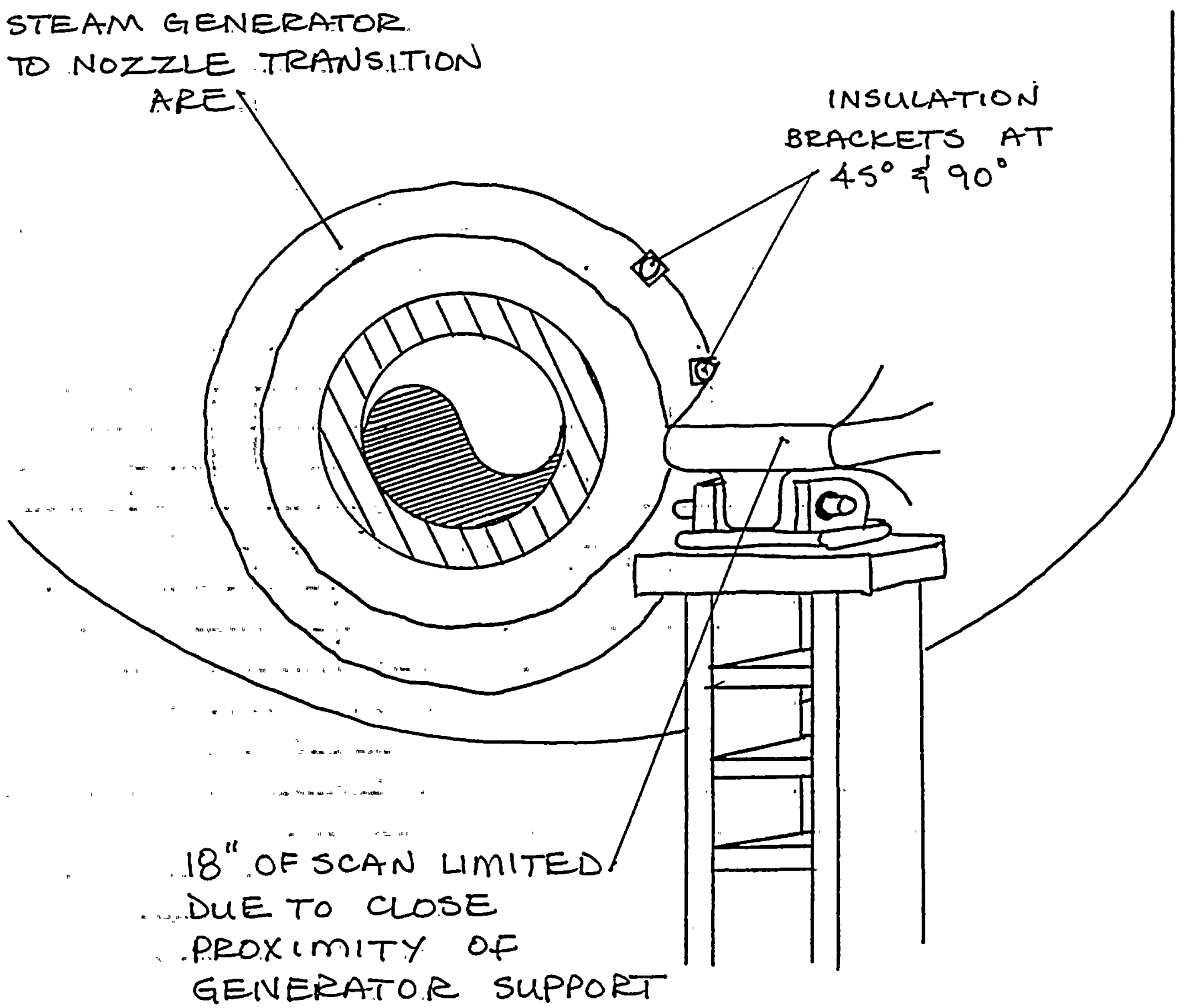
STEAM GENERATOR
"A"
OUTLET NOZZLE
LIMITATIONS




BEST EFFORT EXAM. ESTIMATE 75%
CRV ACHIEVED.

94-4-1 Sheet 1 of 2

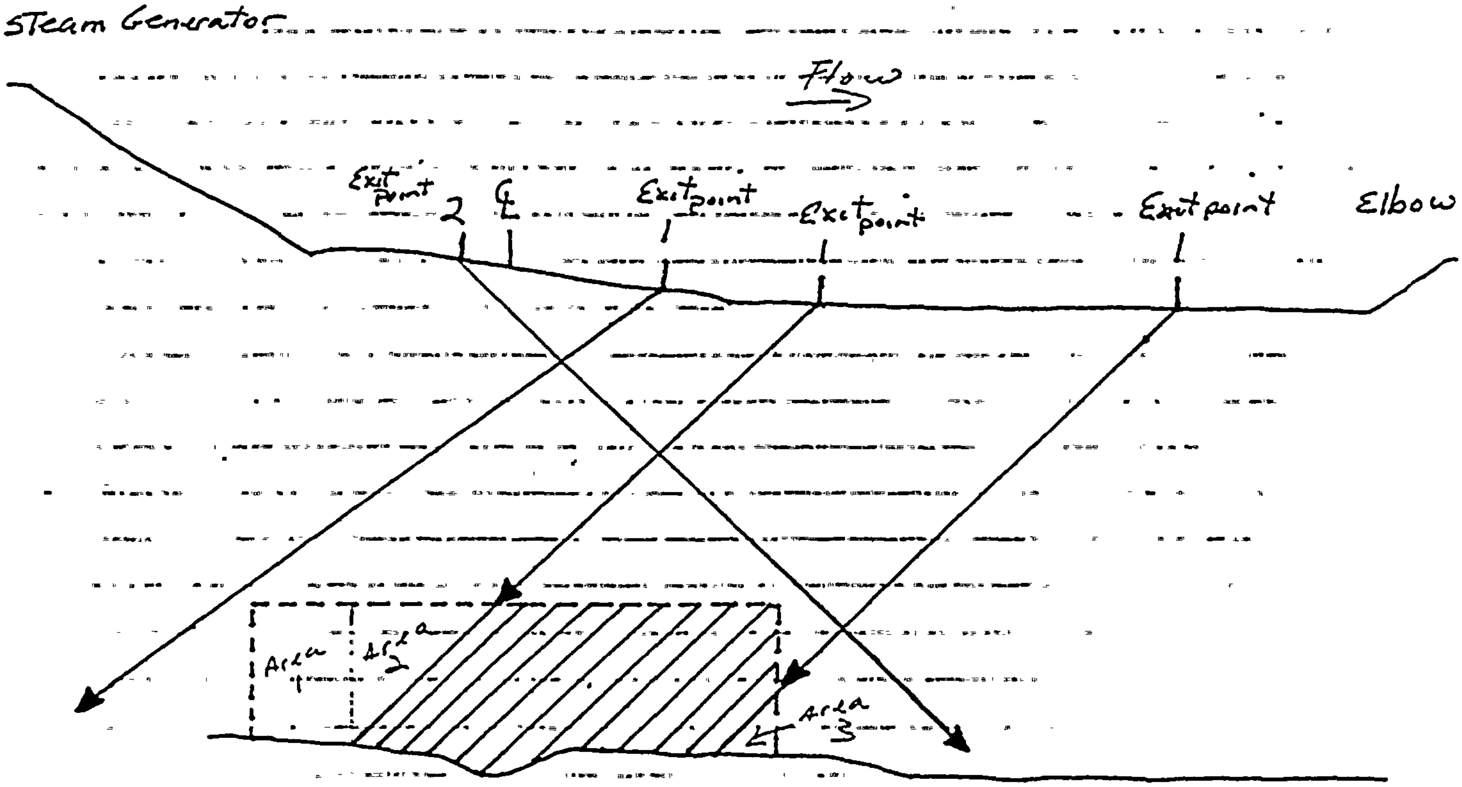
STEAM GENERATOR
A
INLET NOZZLE
LIMITATIONS




BEST EFFORT EXAM. ESTIMATE 75%
CRV ACHIEVED.

94-4-1 Sheet 2 of 2

45° CRV 31" RCS-1401-5




Total Area = 1.89

Area #1
w = .364

Area #2
½ b·H = .20

Area #3
½ b·h = .093

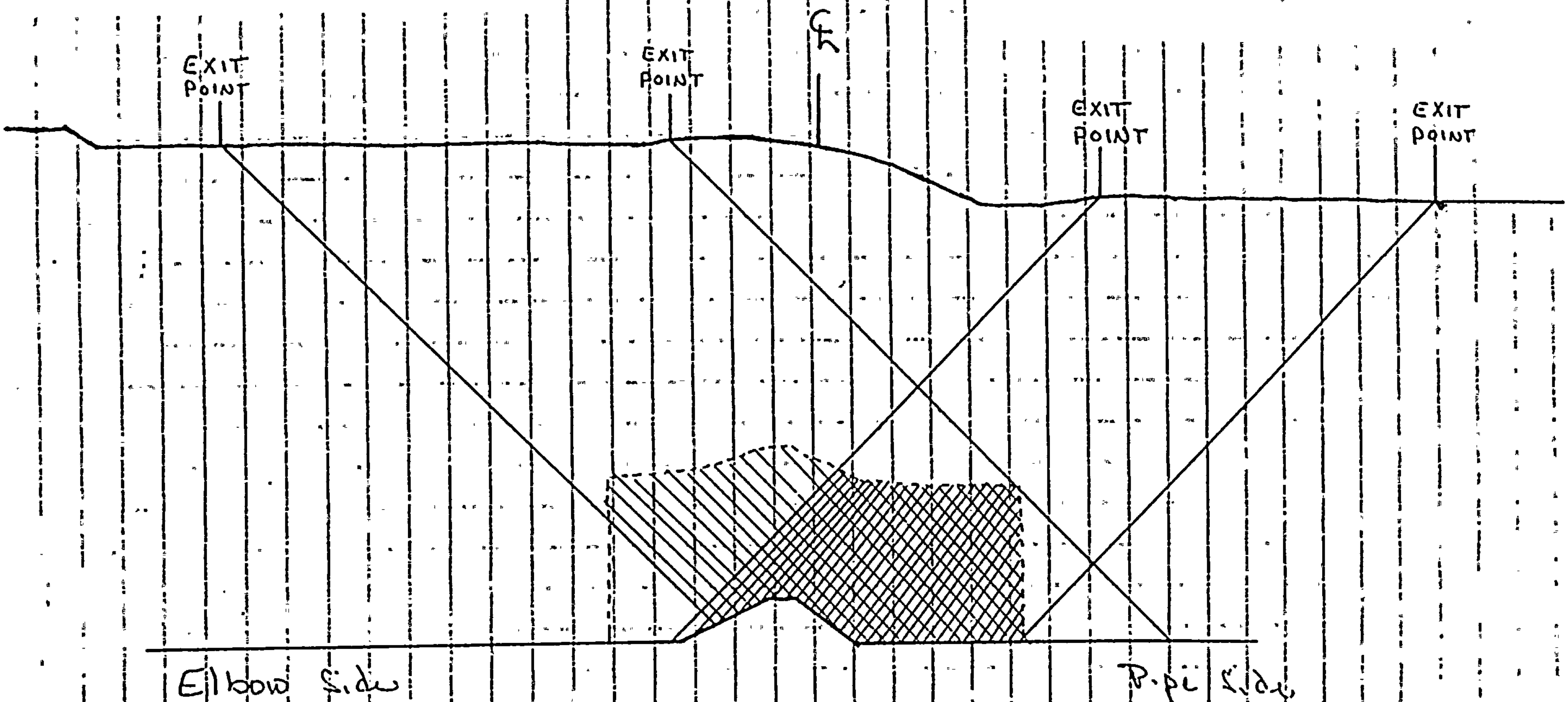
No coverage @ wall = .72
Area scanned in one direction = 1.17

Upstream Side Coverage = 0%

Downstream Side Coverage = 62%
ONE SIDED EXAMINATION

94-4-2

31"-RCS-1401-8

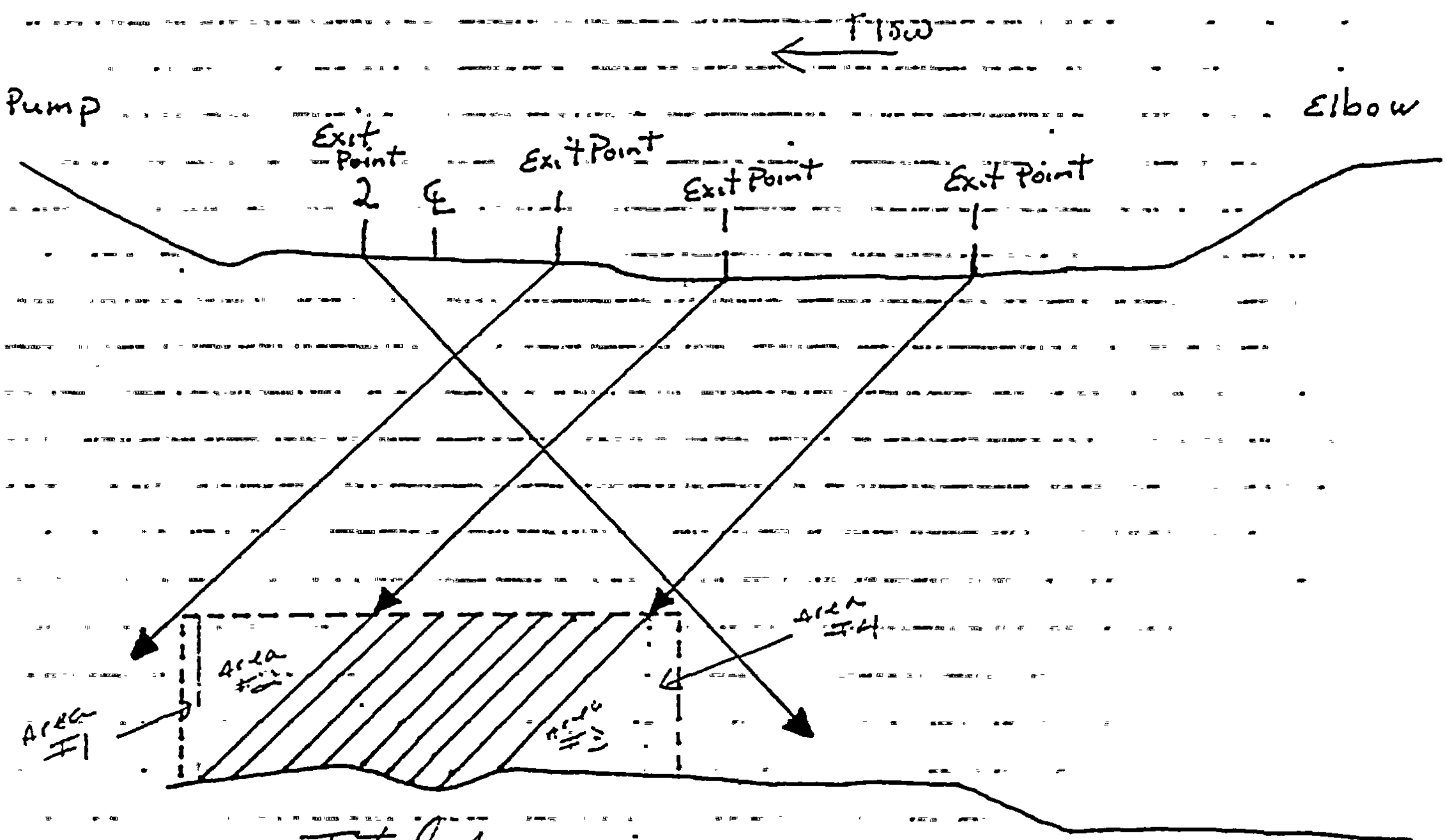



45° 1/2 V
CRV

64% CRV achieved from pipe side
88% CRV achieved from elbow side

31"-RCS-1401-10

4.5° CRV




Total Area = 2.10

| Area #1 | Area #2 | Area #3 | Area #4 |
|---|---|---|---|
| L×W = .077 | ½ b·h = .37 | ½ b·h = .34 | L×W = .128 |

No coverage @ all = .92

Area scanned in 1 direction = 1.24

Upstream Side Coverage = 57%
ONE SIDED EXAMINATION
Down Stream Side = 0%

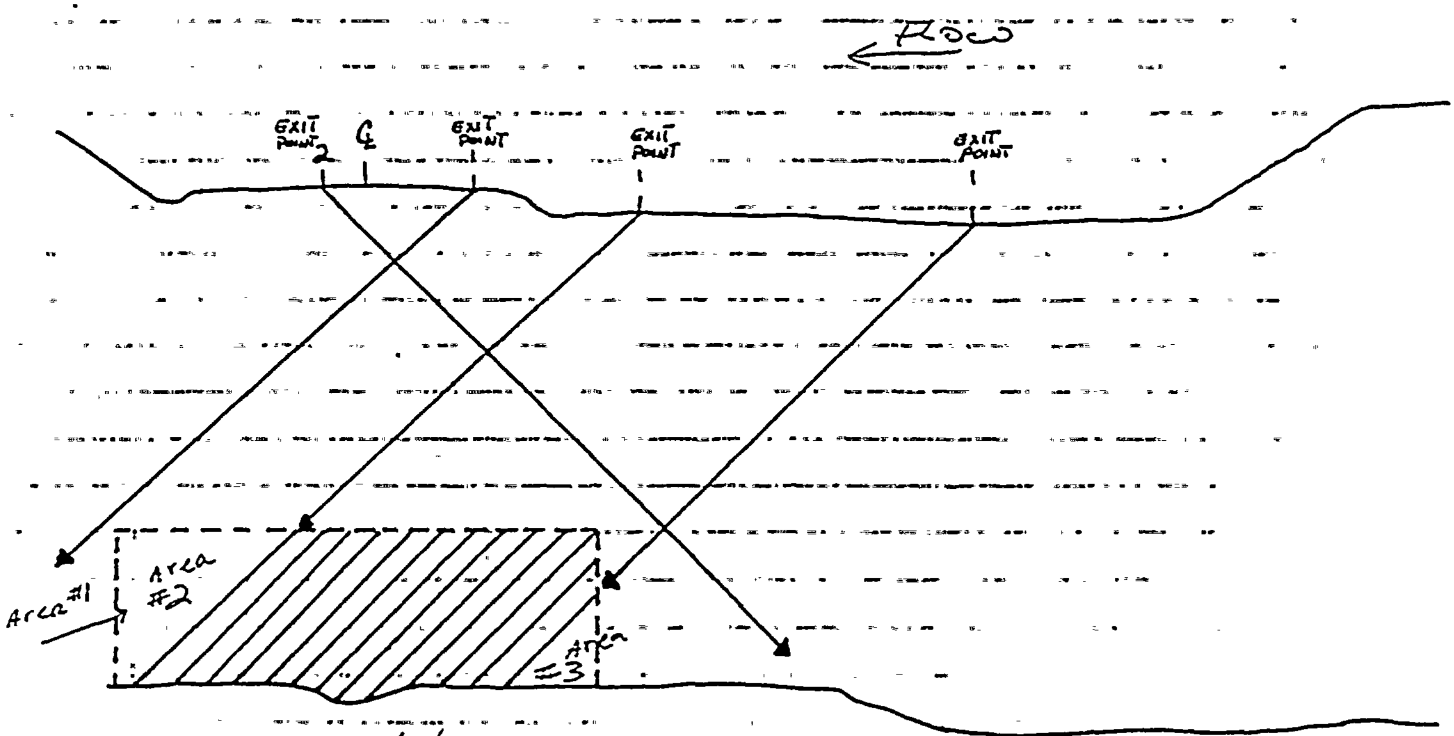
94-4-4

45° CRV
1/2 V

Zone 4-008

29"-RCS-1404-4

STEAM GENERATER ELBOW



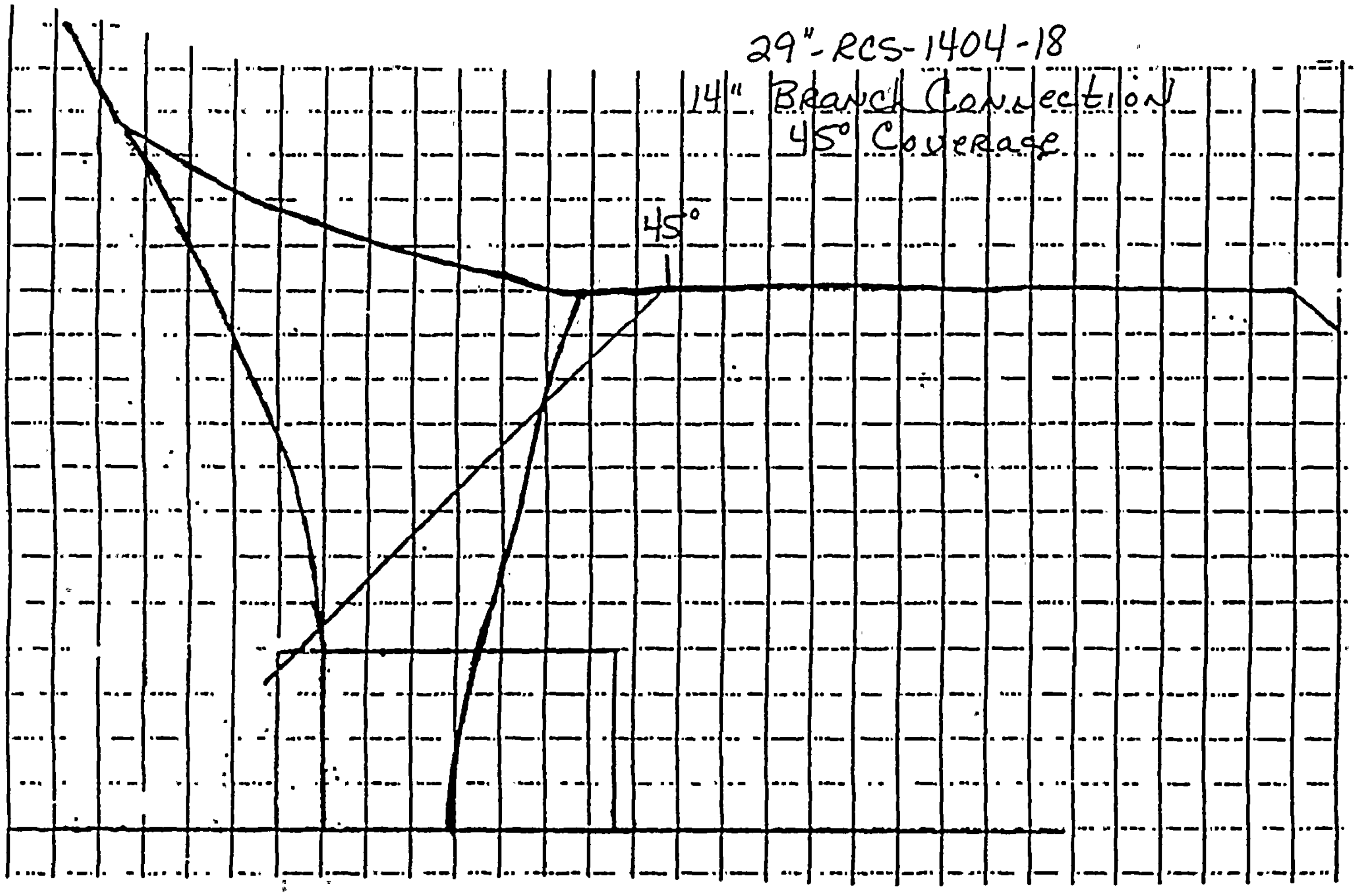

Total Area = 2.05

Area #1 Area #2 Area #3
L X W = 0.82 1/2 b·h = .323 1/2 c·h = .113

No coverage @all = .523
Area scanned in 1 direction = 1.527

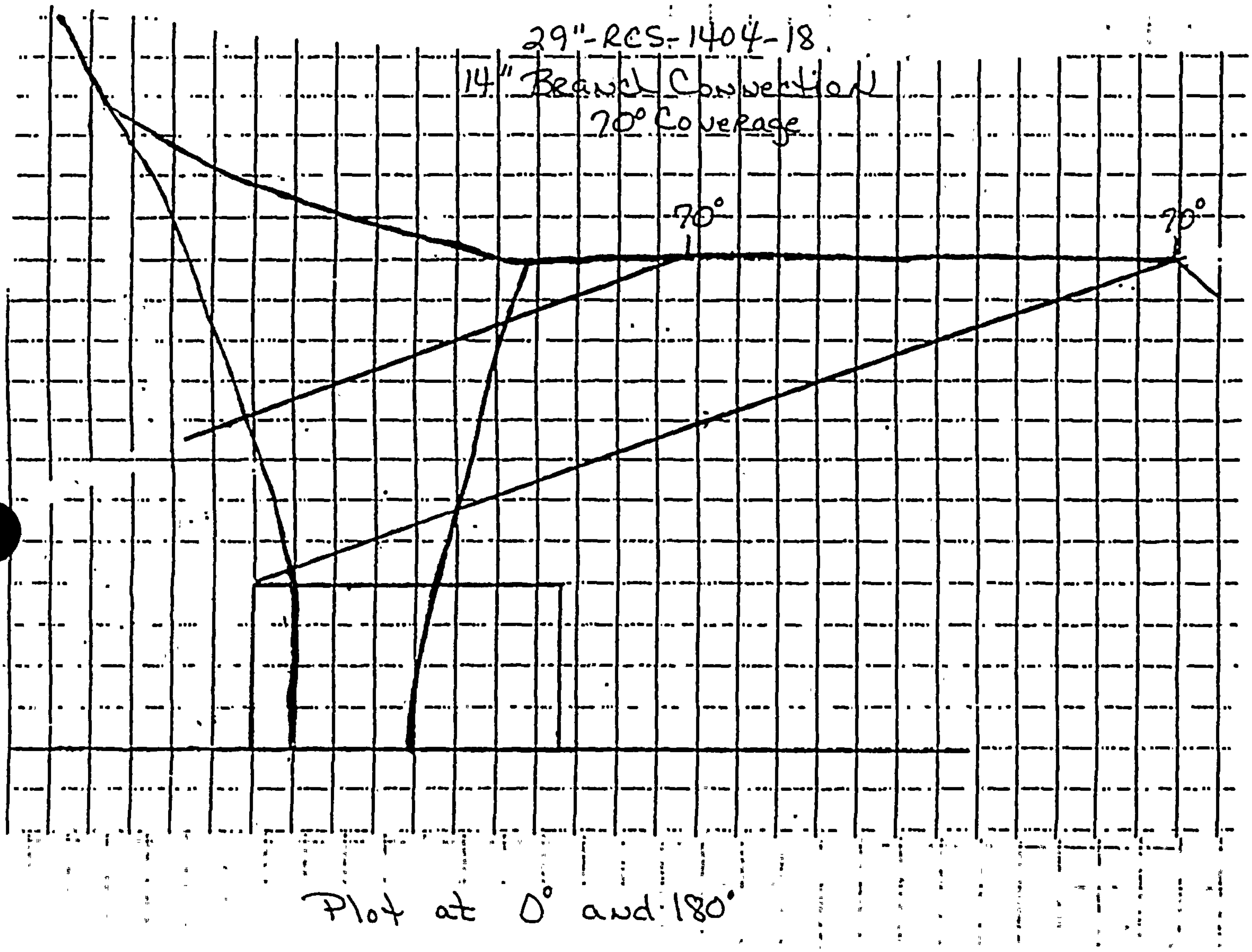
Up Stream Side Coverage = 74%
ONE SIDED EXAMINATION
Down Stream Side Coverage = 0%

94-4-5

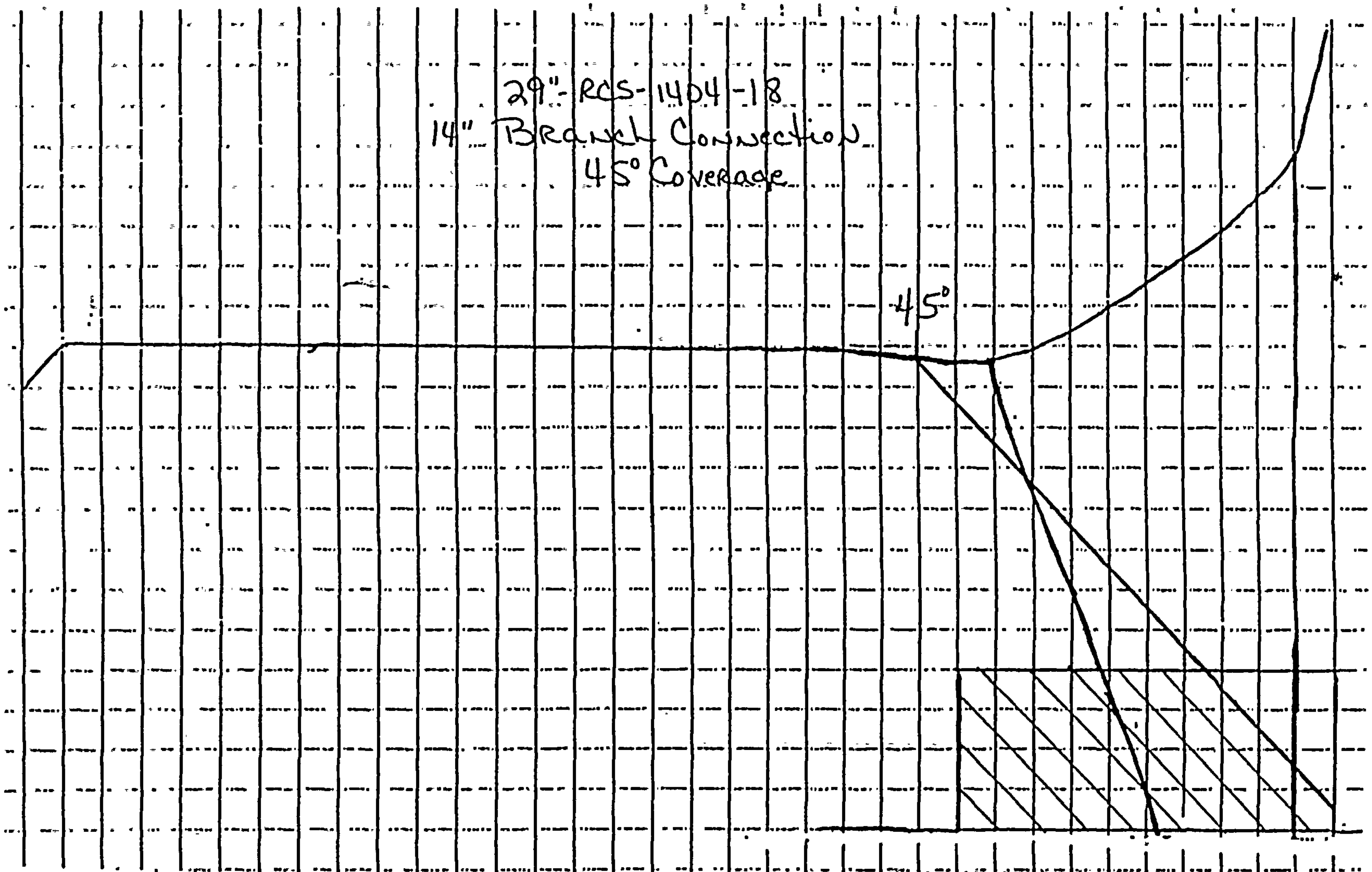



Plot at 0° and 180°

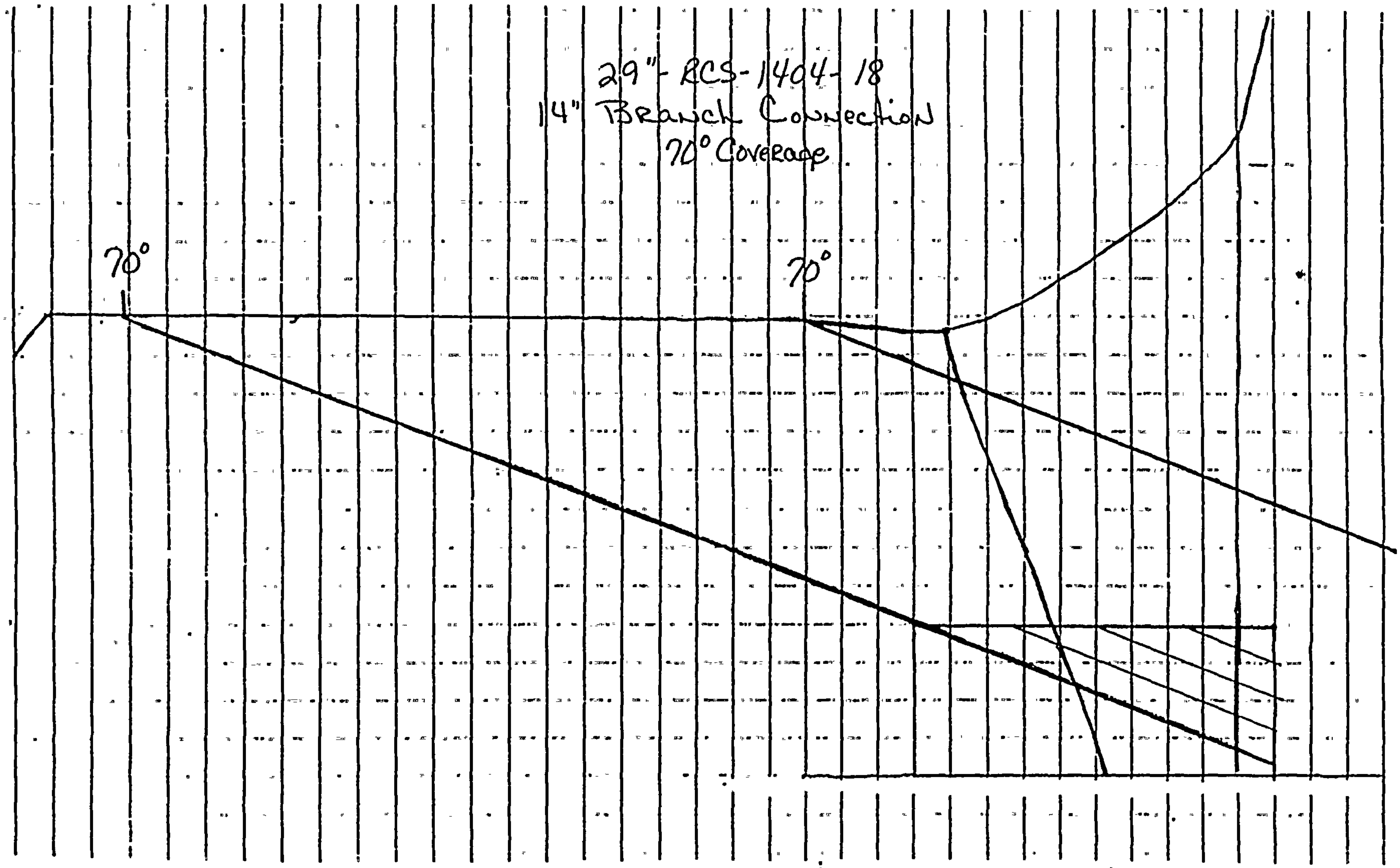
94-4-6 Sheet 1 of 4




94-4-6 Sheet 2 of 4




Plot at 90° + 270°



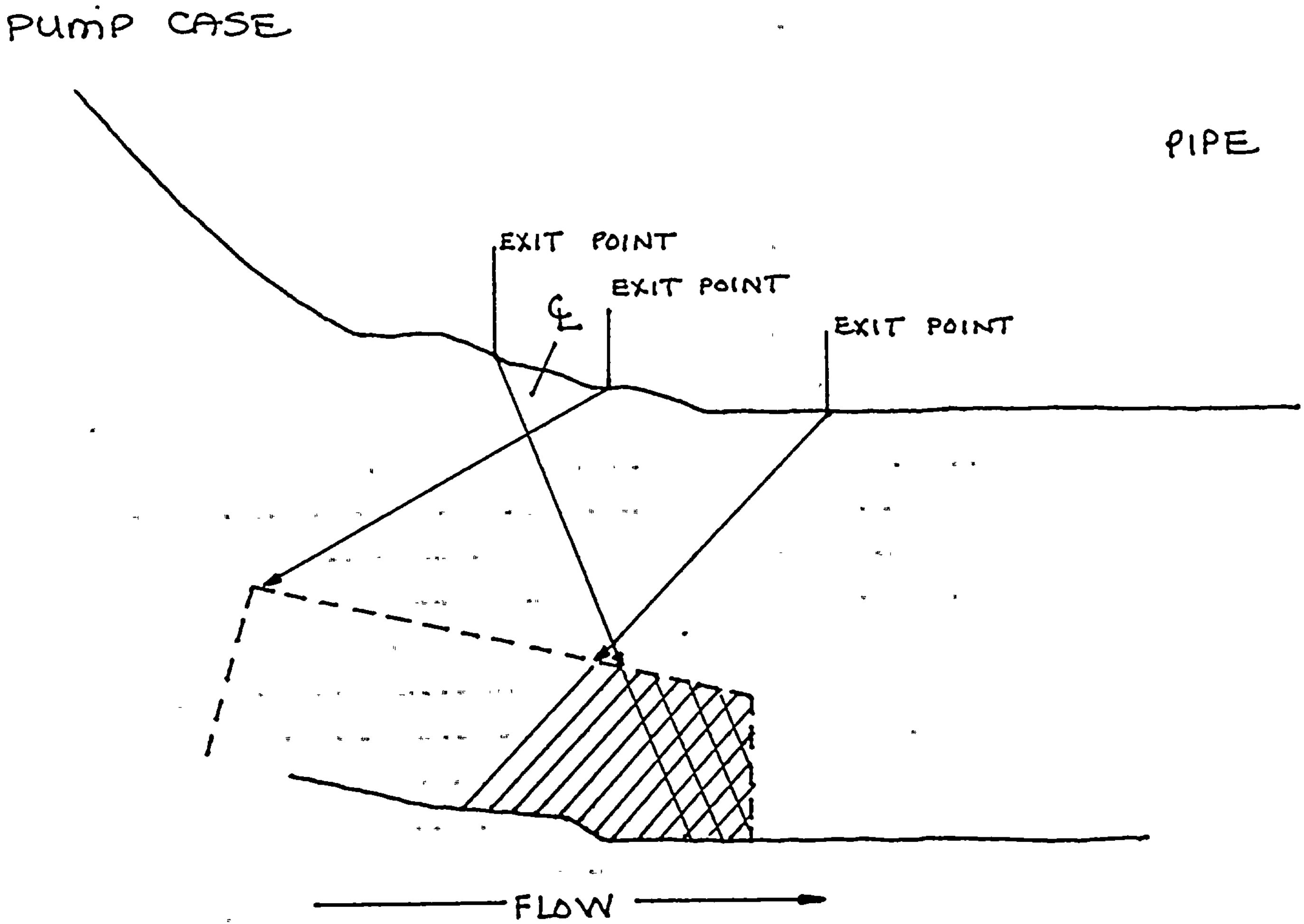

CRV

Plot at 90° & 270°

100% CRV from Branch Side
0% CRV from Main Lap Side

LOOP A COLD LEG
27.5" RCS-1407-11

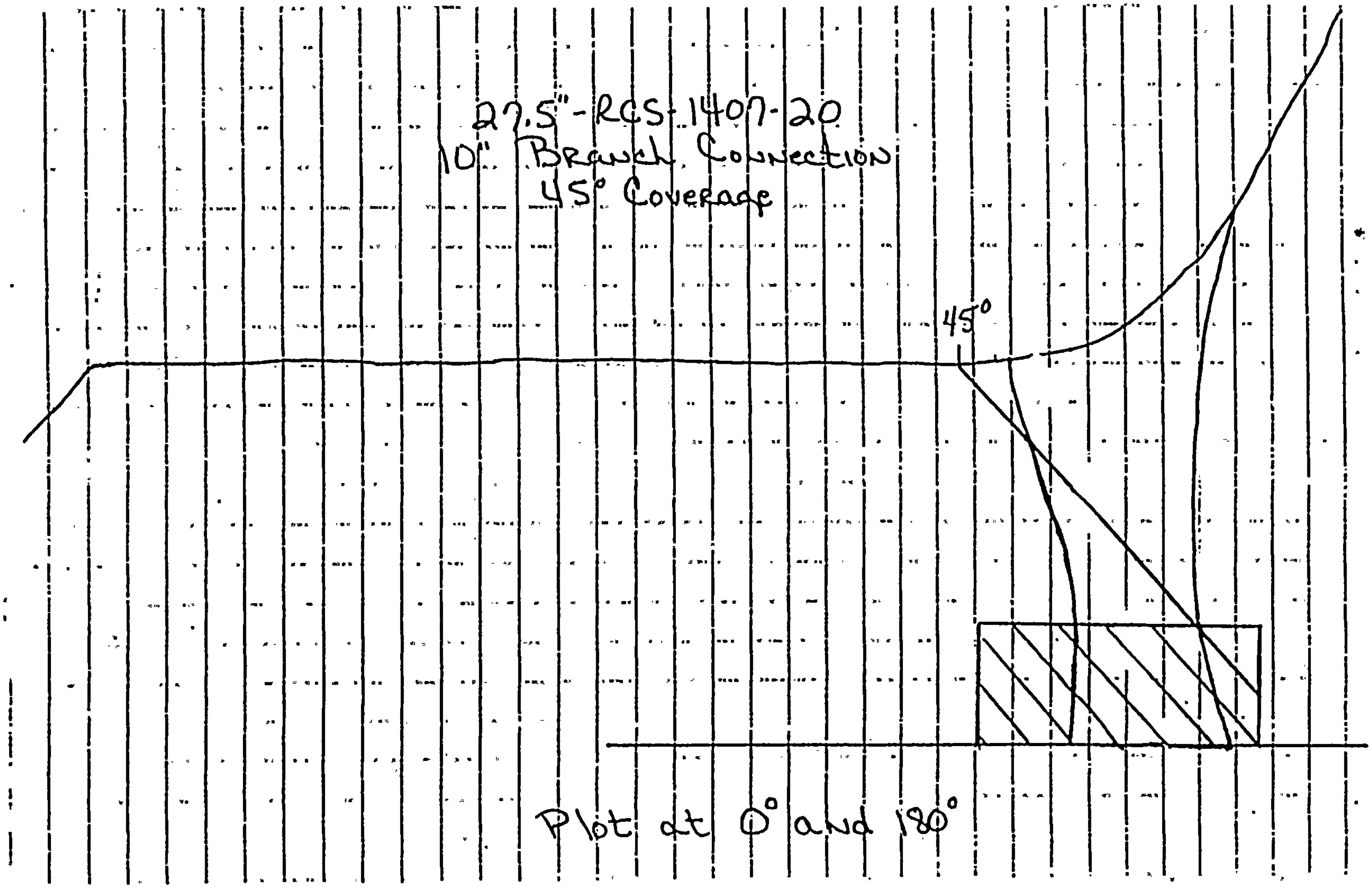
45° CRV
1/2 V



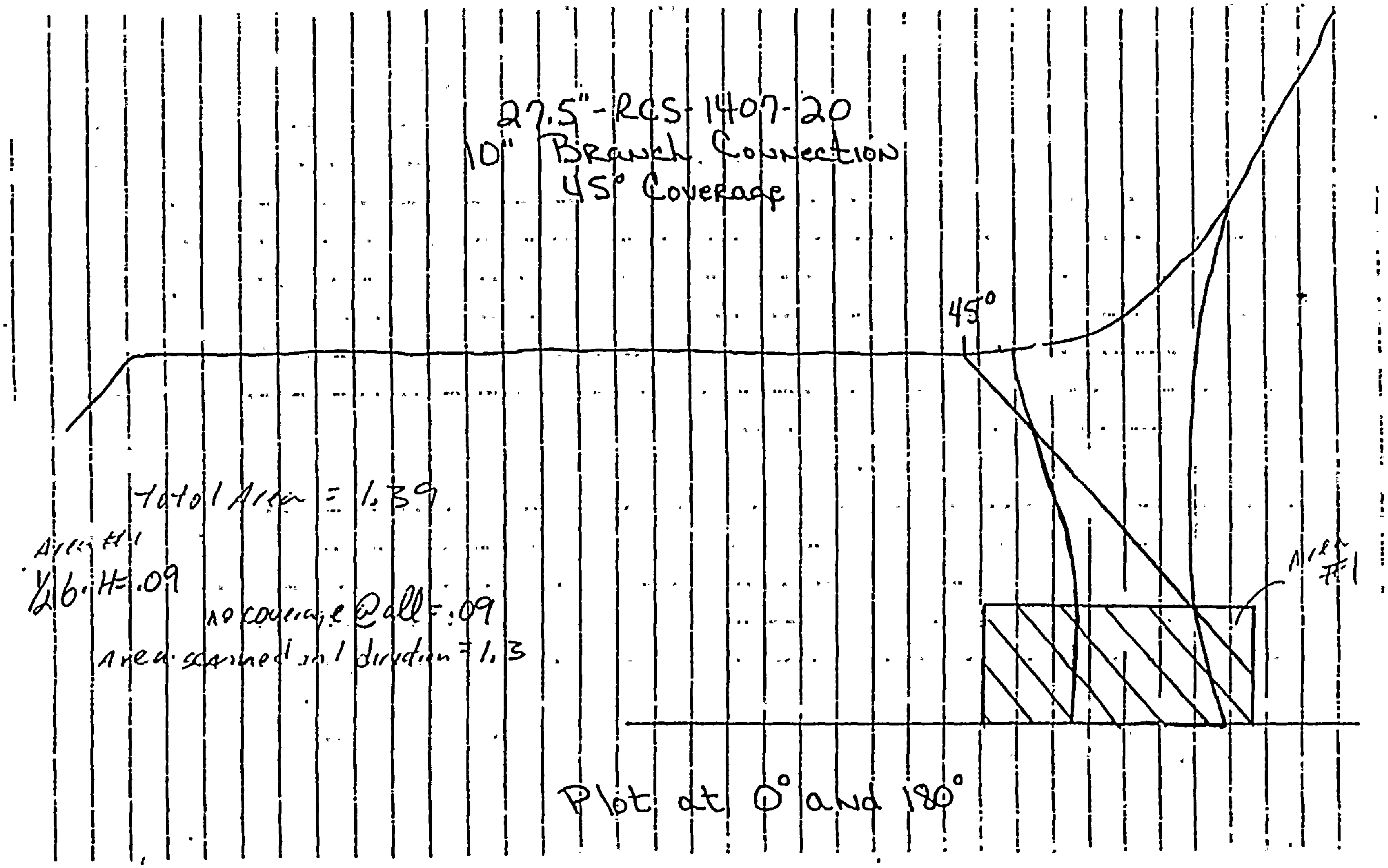

18% CRV achieved from Pump Side —
41% CRV achieved from Pipe Side

ONE SIDED EXAM

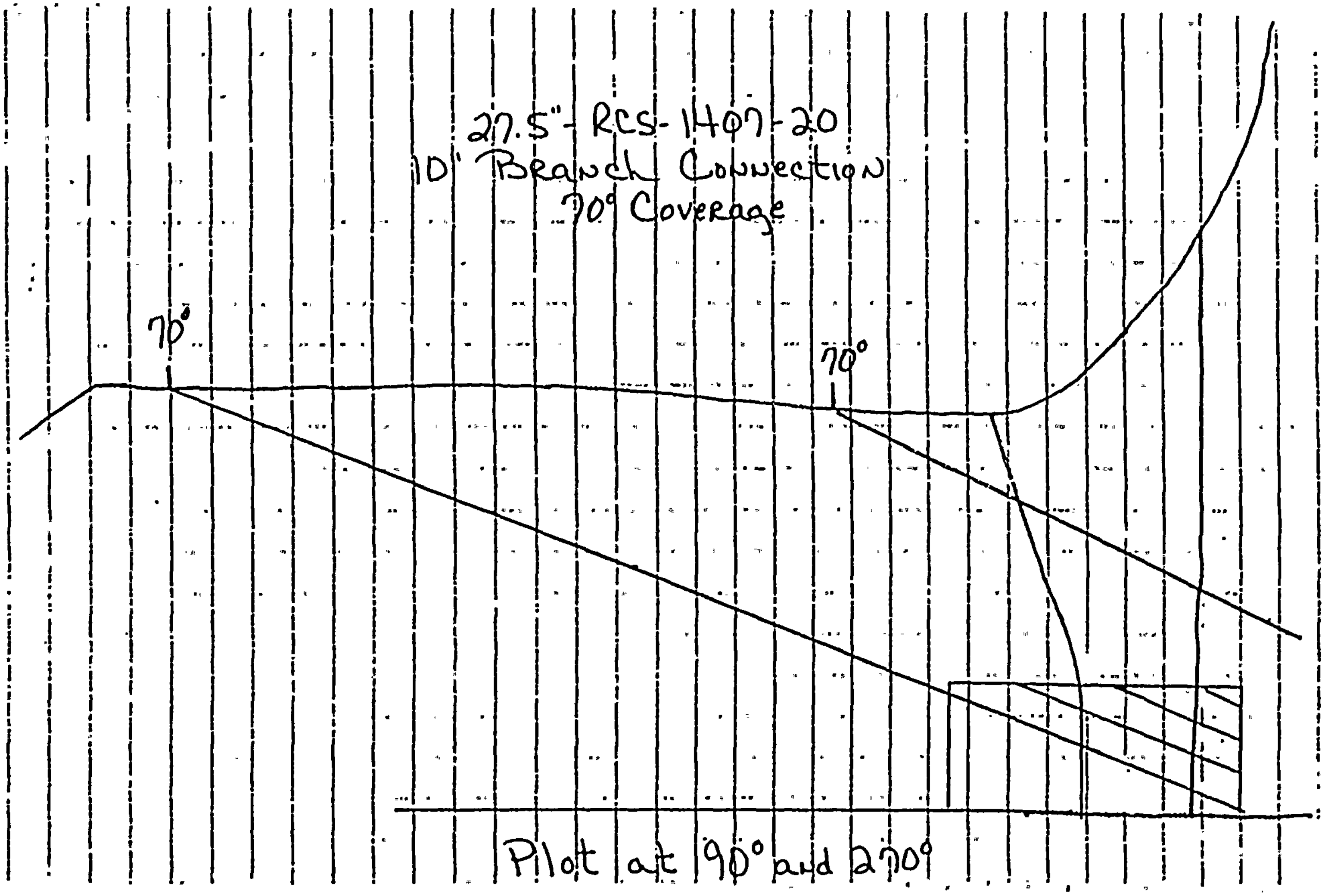
94-4-7



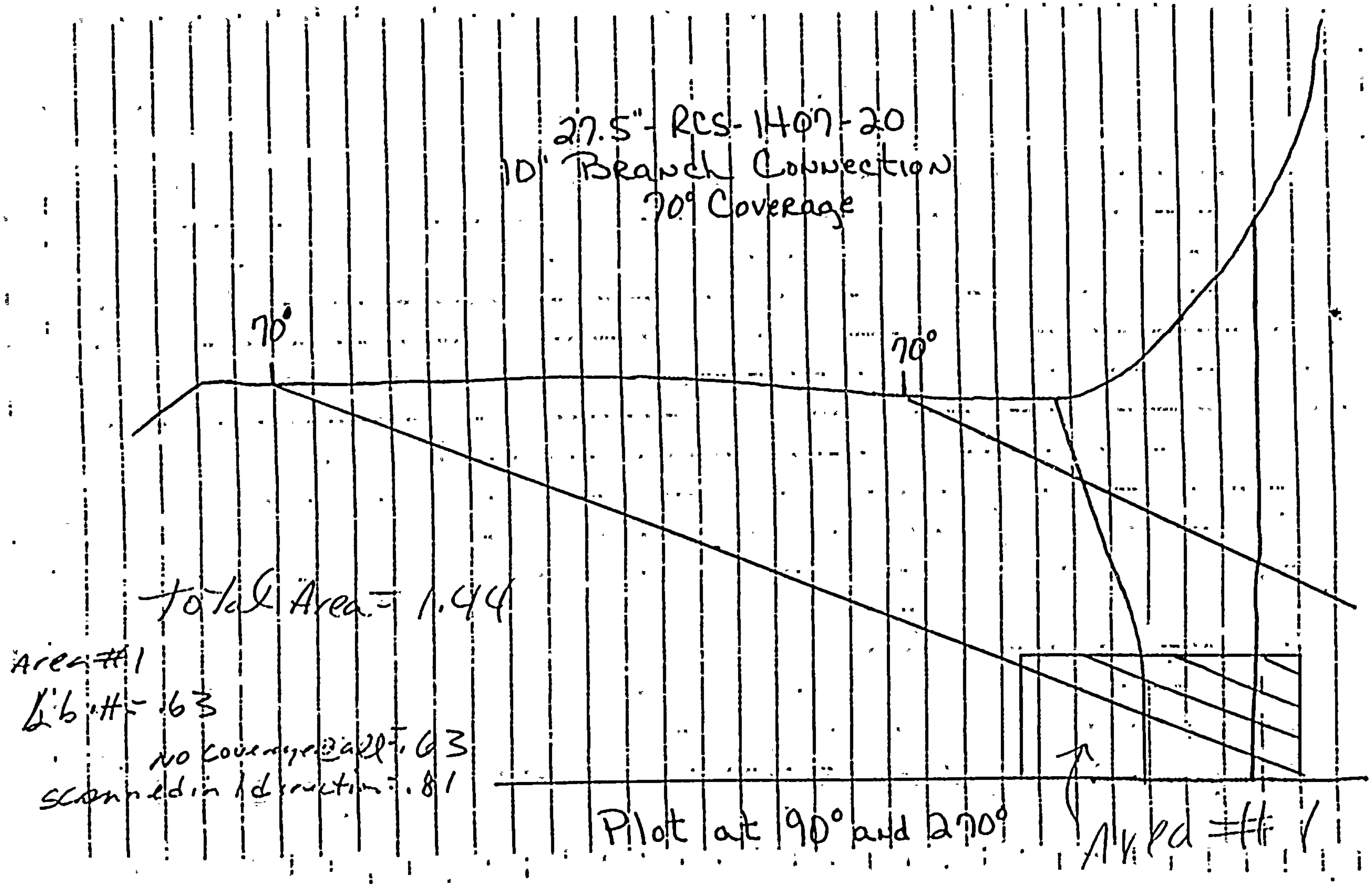

94-4-8 Sheet 1 of 8




UpStream Coverage = 94%
Downstream Coverage 0%

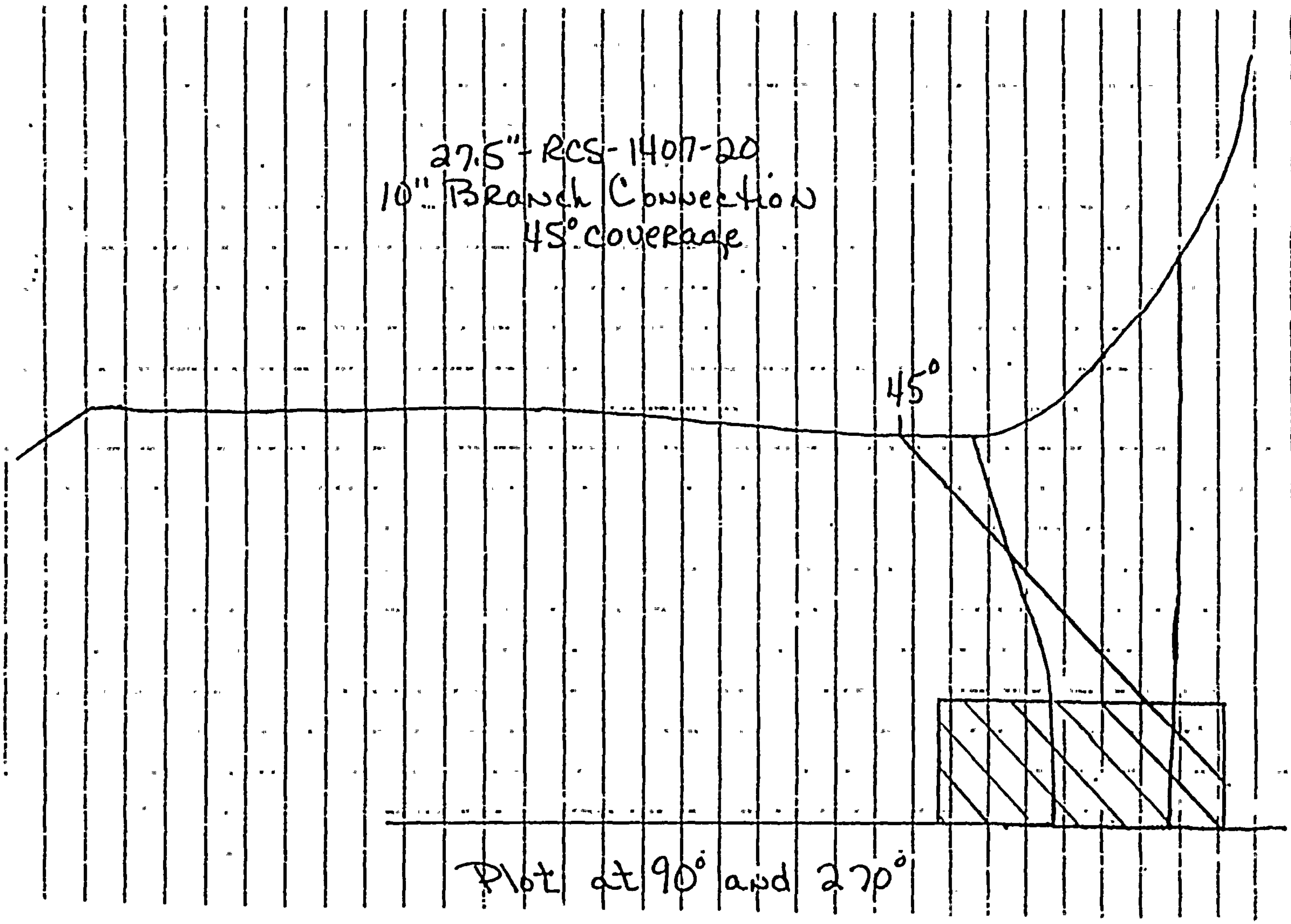


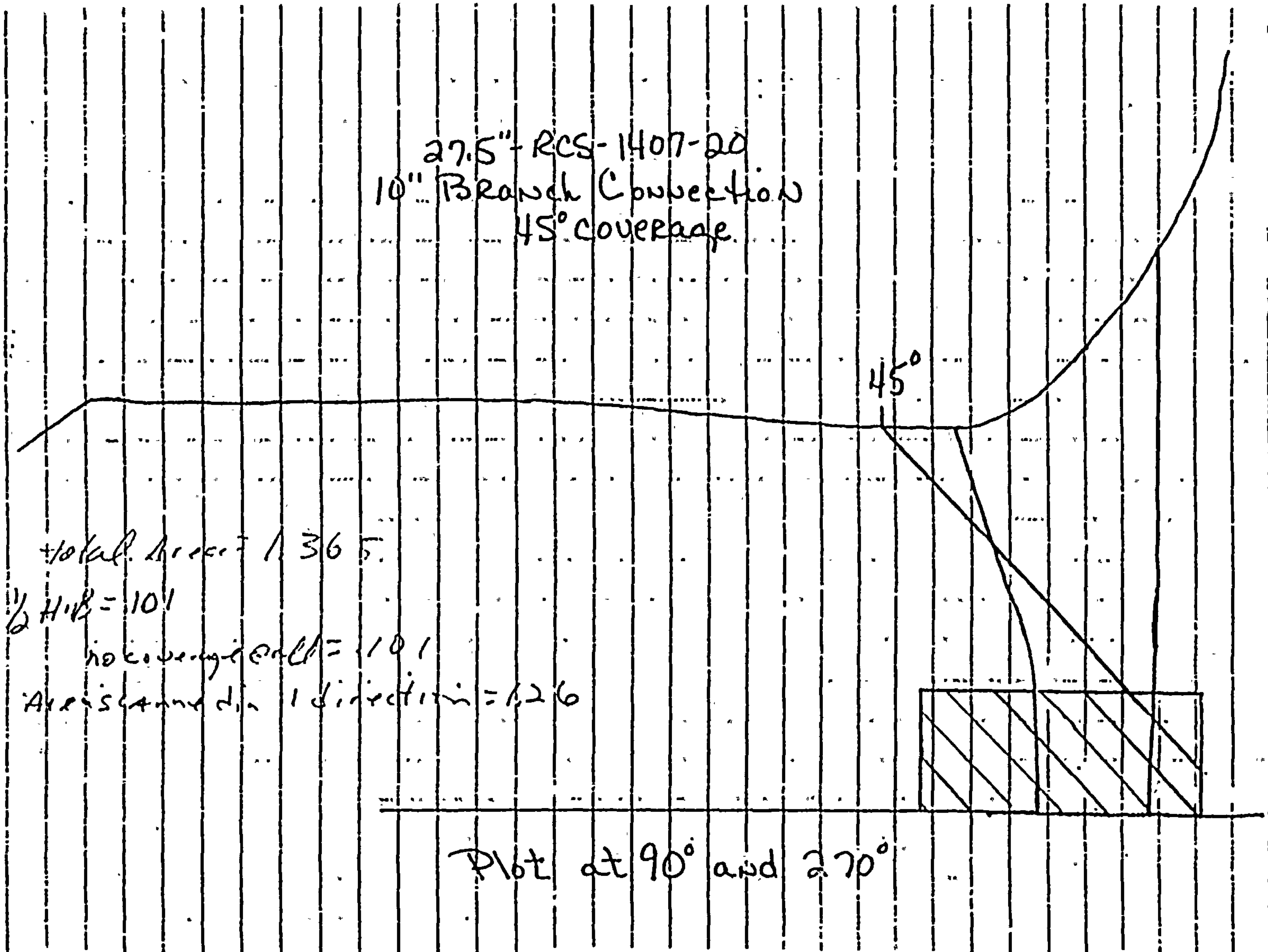

94-4-8 Sheet 3 of 8




Upstream Side : 56%

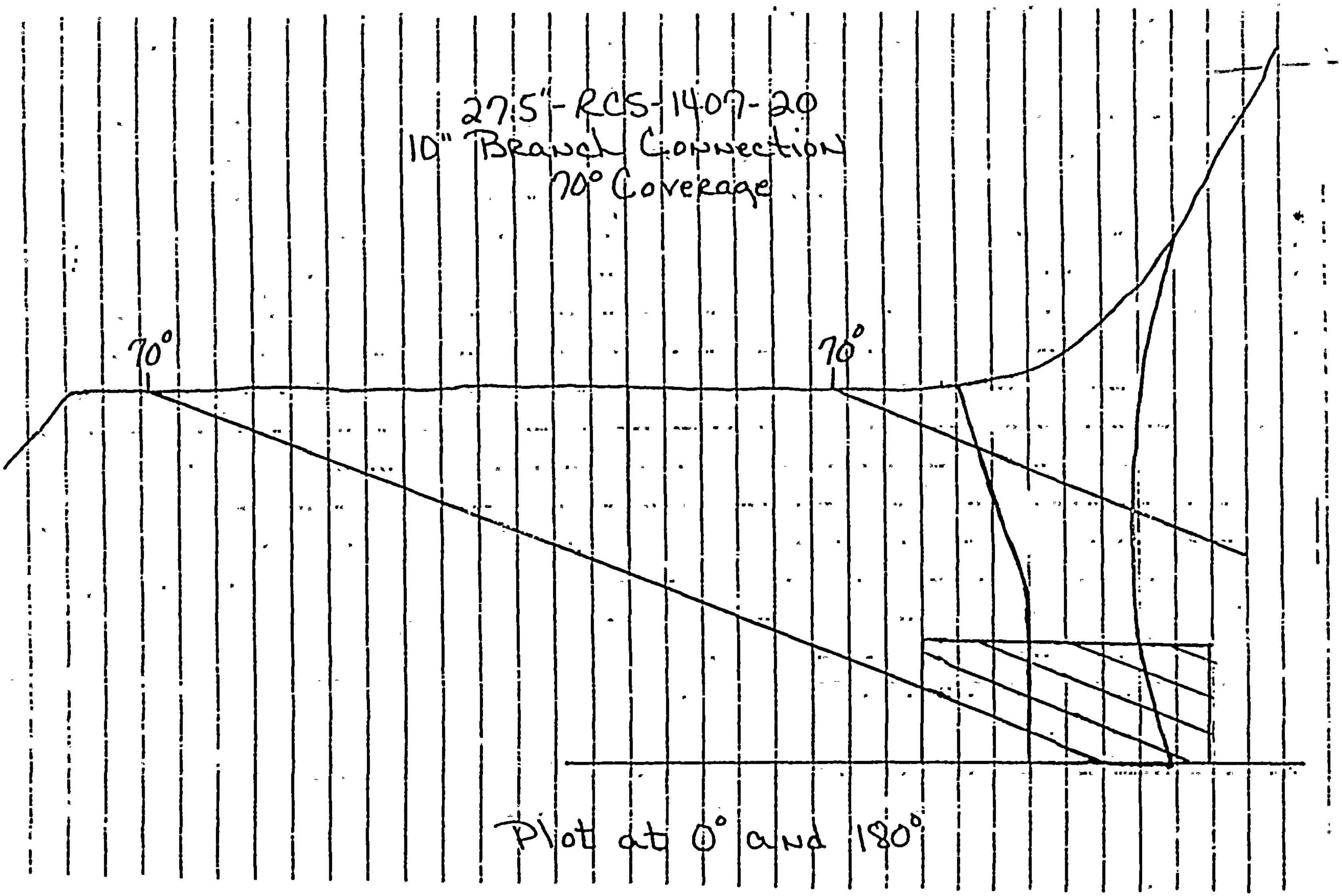
Downstream Side : 0%


27.5"-RCS-1407-20
10" Branch Connection
45° coverage
45°
Plot at 90° and 270°



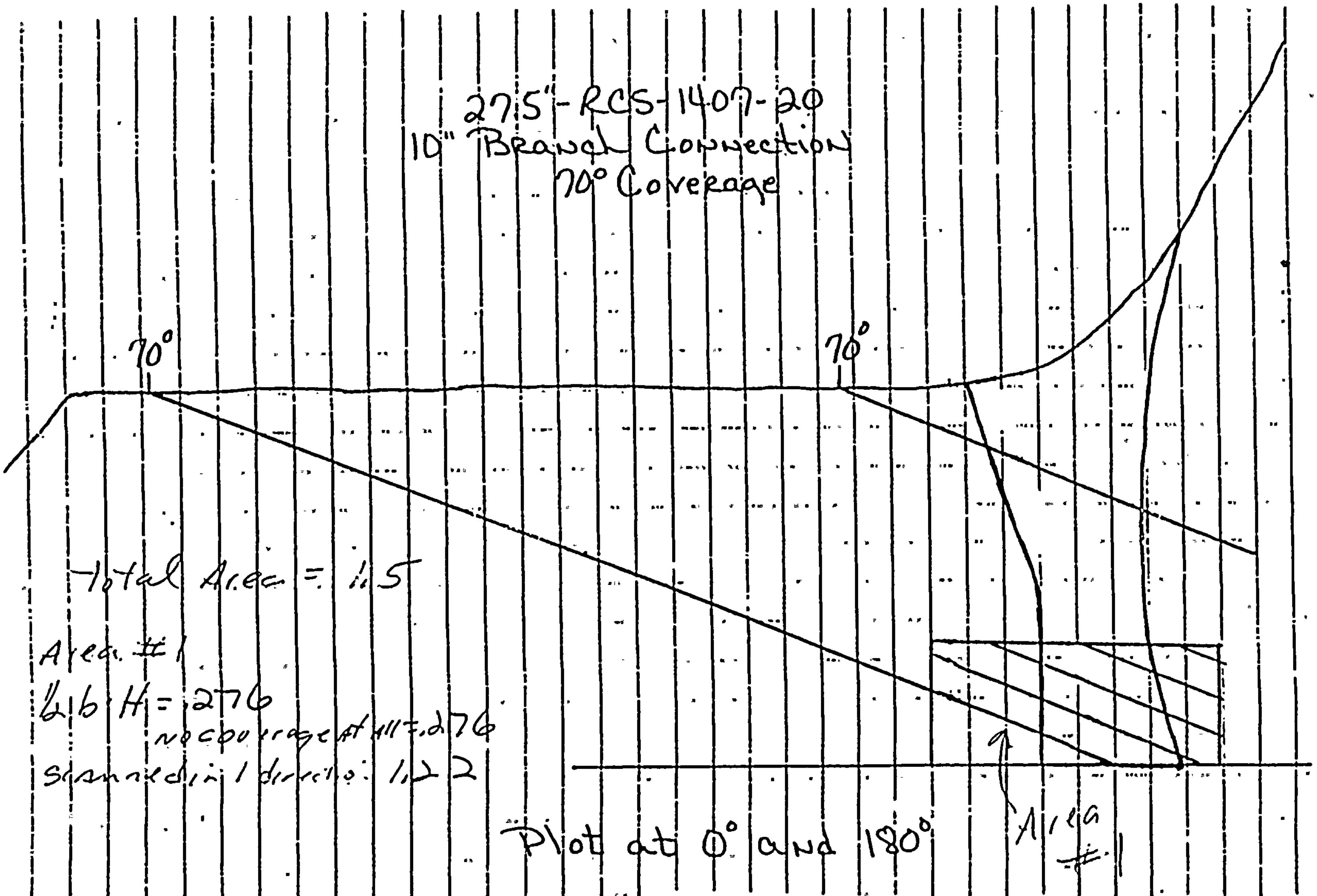

up stream Coverage = 92.0%
Downstream Coverage = 0%

23

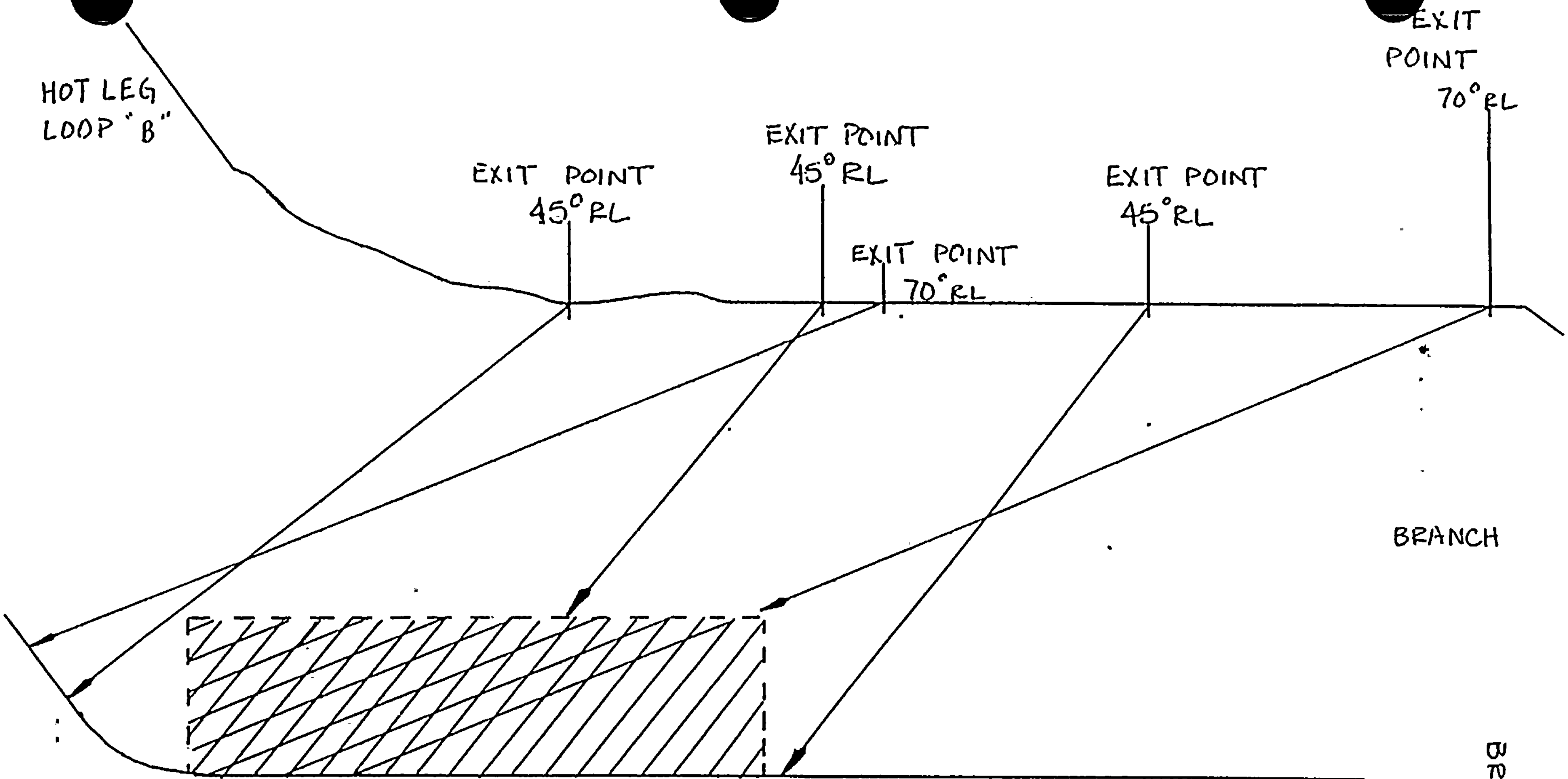



24

94-4-8 Sheet 7 of 8




Upstream Coverage = 81%
Down Stream Coverage = 0%



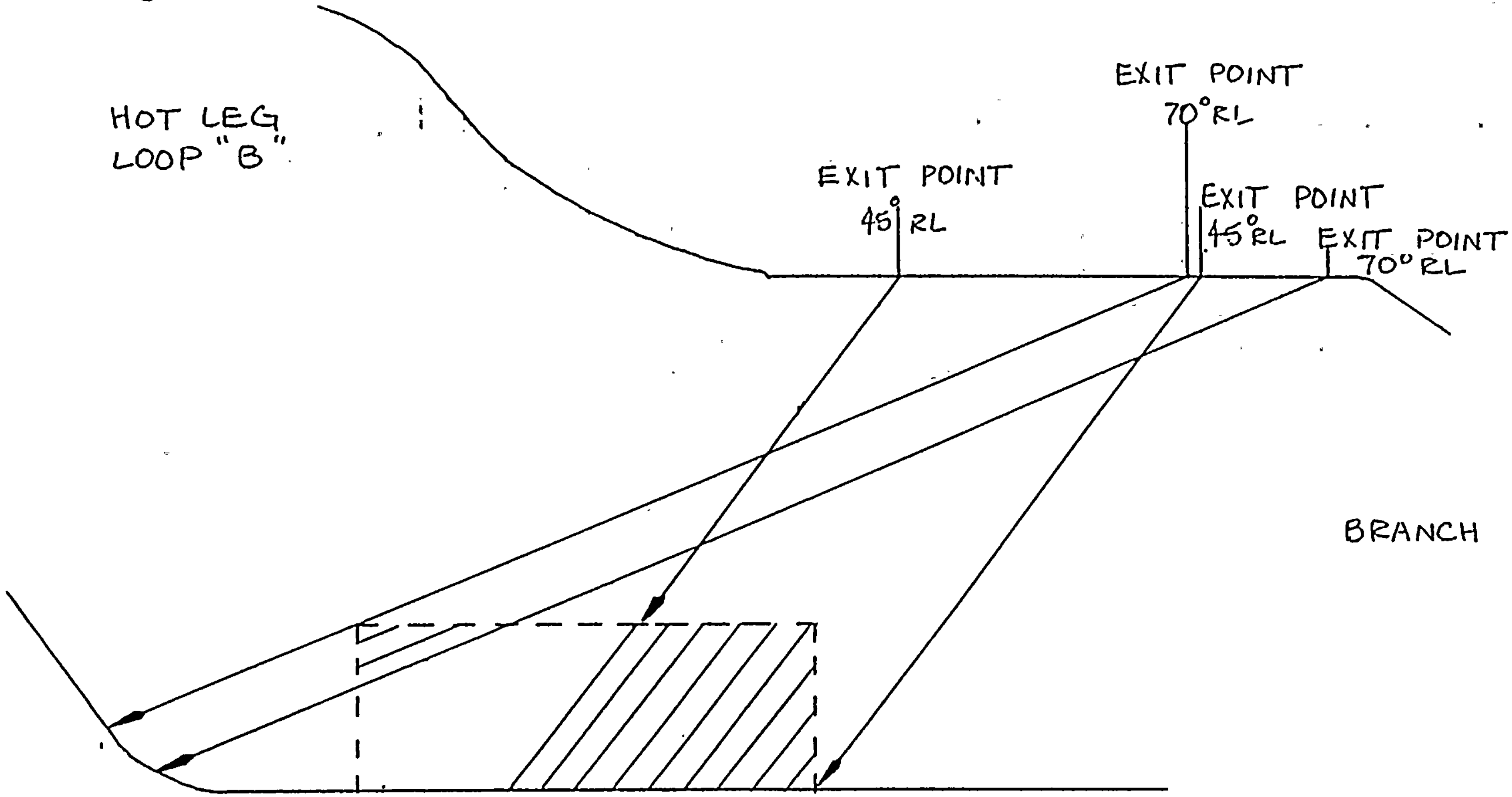

WELD CROWN WIDTH – 3.3"
CIRCUMFERENCE – 52.0"

45° RL AND 70° RL
CRV
1/2 V
LANDING AT 0° AND 180° – 5.2"

29" RCS-1405-21
BRANCH CONNECTION TO
12" RCS-1401

94-4-9 Sheet 1 of 2

75% ~~80%~~ 11/2/94 CRV from nozzle side
0% CRV from pipe/branch side

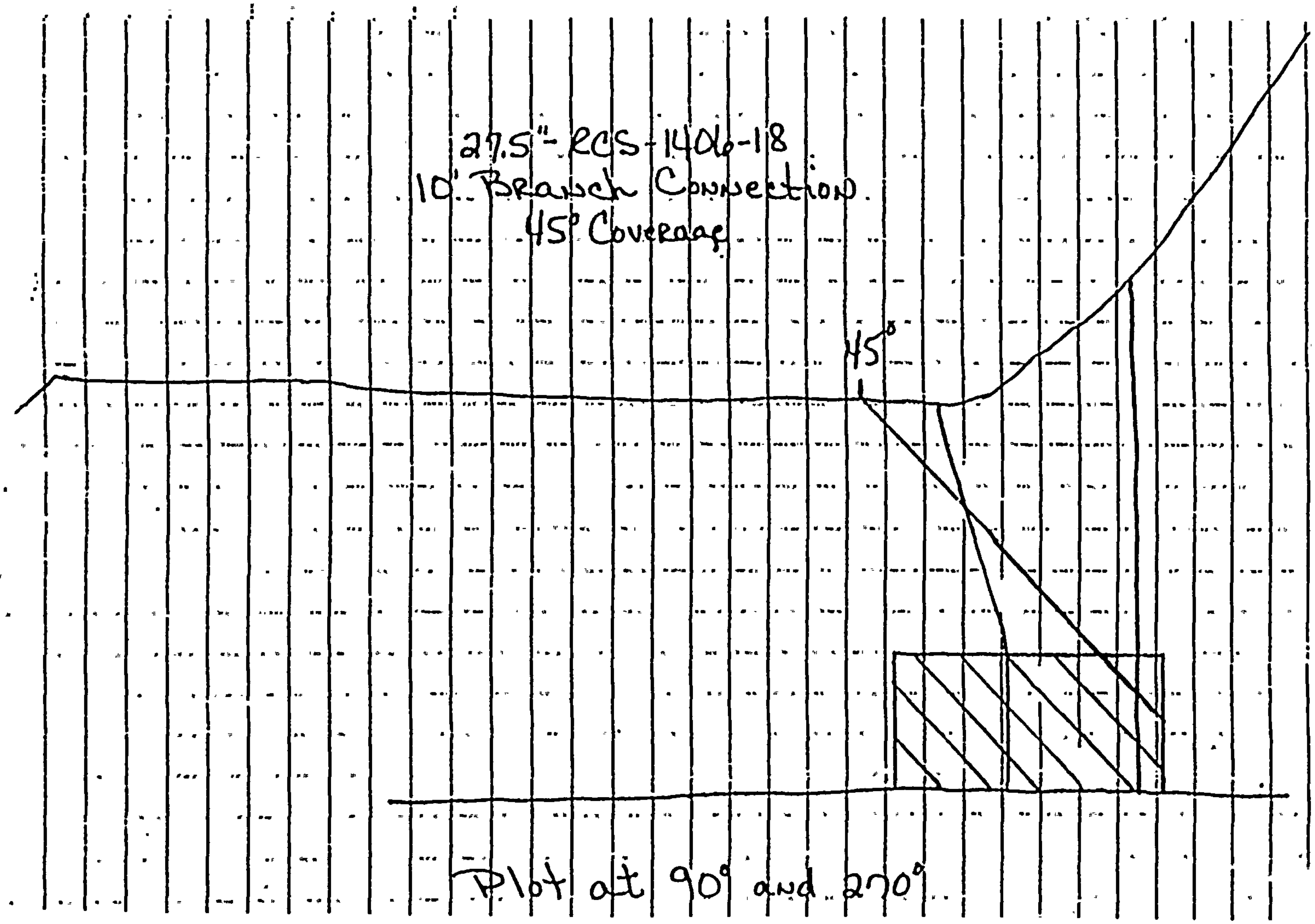



29" RCS-1405-21
BRANCH CONNECTION TO
12" RCS-1401

WELD CROWN WIDTH - 2.7"
CIRCUMFERENCE - 52.0"

45°RL & 70°RL
CRV
1/2 V
LANDING AT 90° & 270° - 3.7"
See previous page for CRV coverage

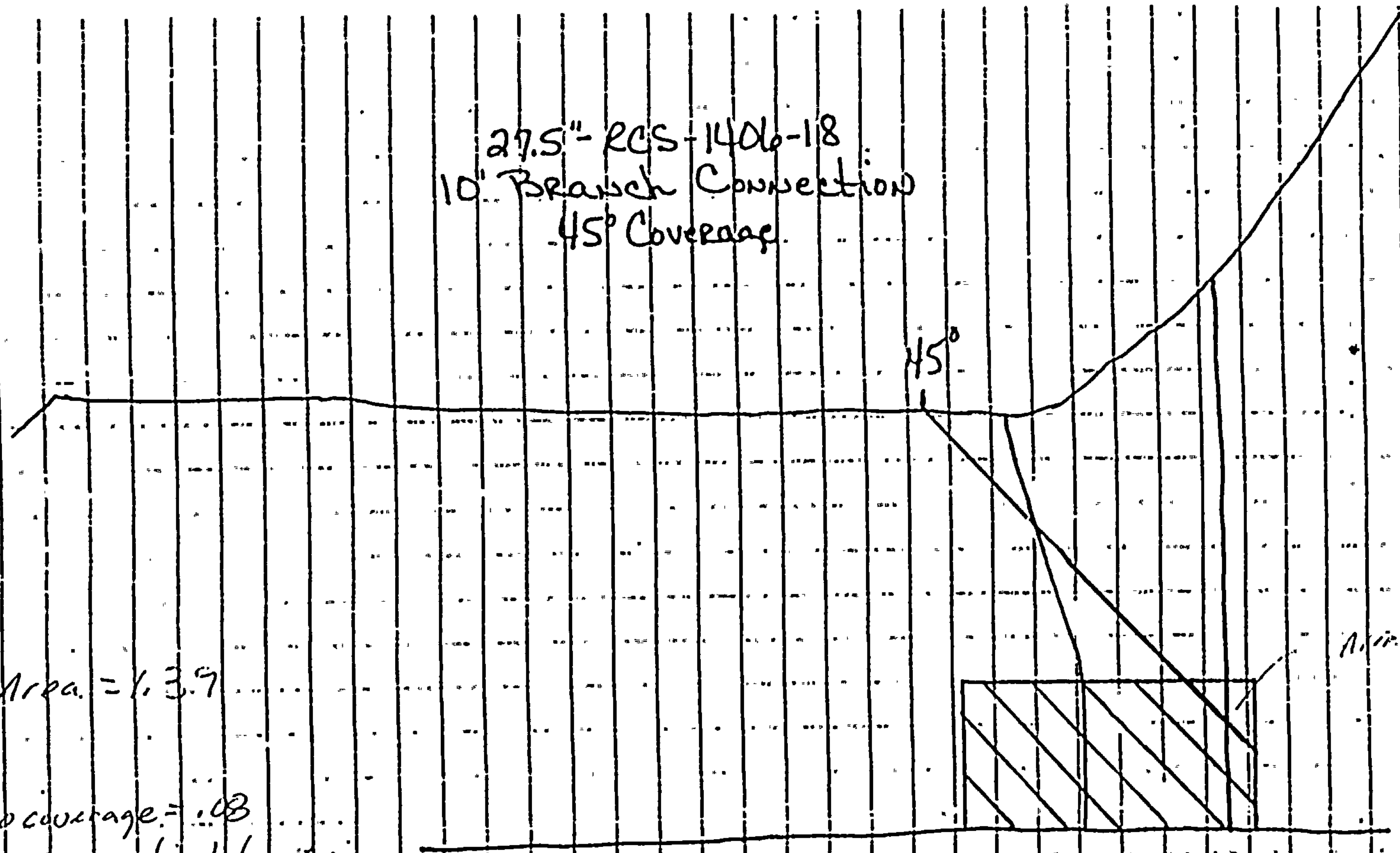
94-4-9 Sheet 2 of 2






94-4-10 Sheet 1 of 8

27.5" - RCS-1406-18
10" Branch Connection
45° Coverage




Total Area = 1.39

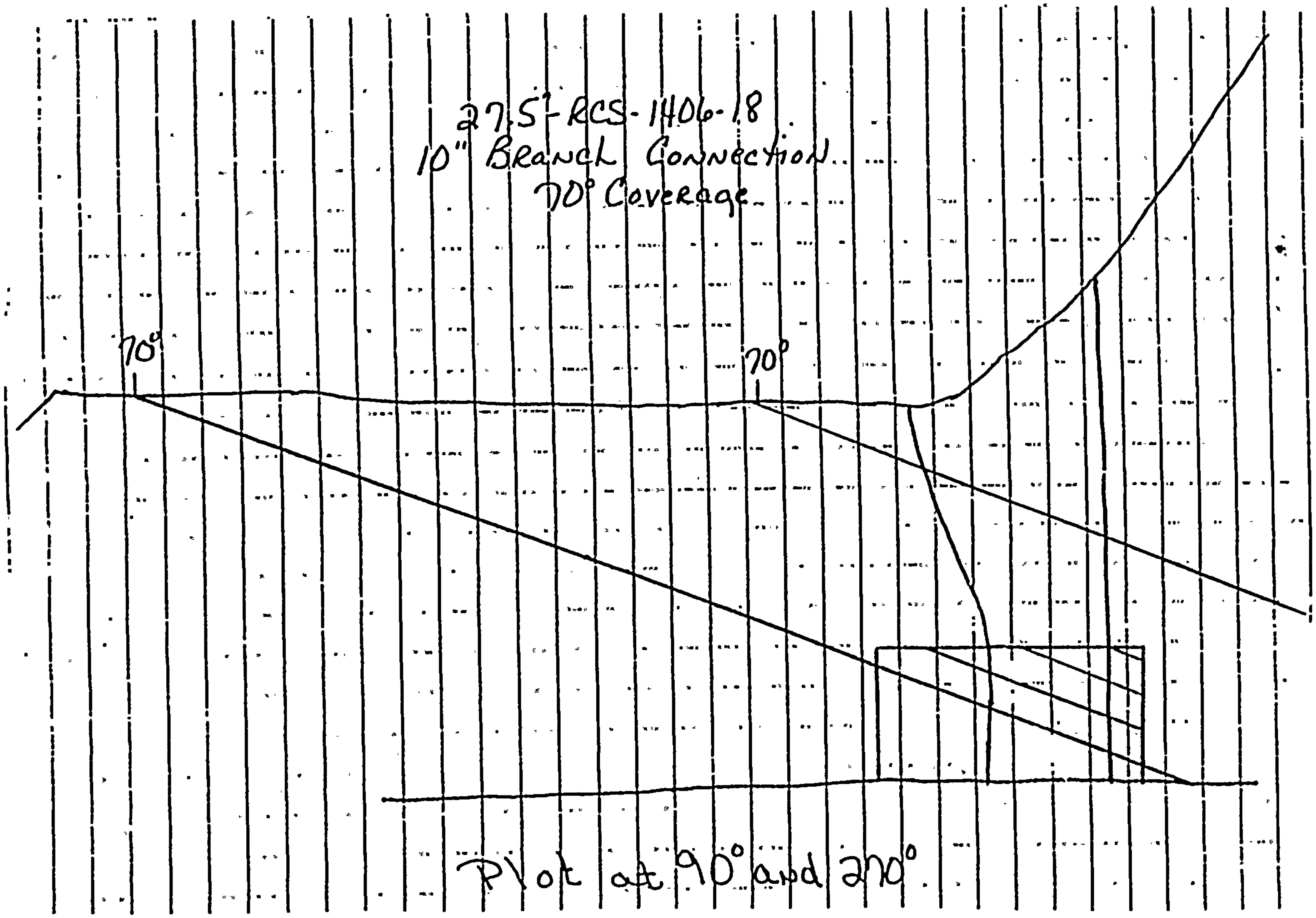
Area #1
1/2 .4 .4 = .08

no coverage = .08
Area scanned in 1 direction = 1.31

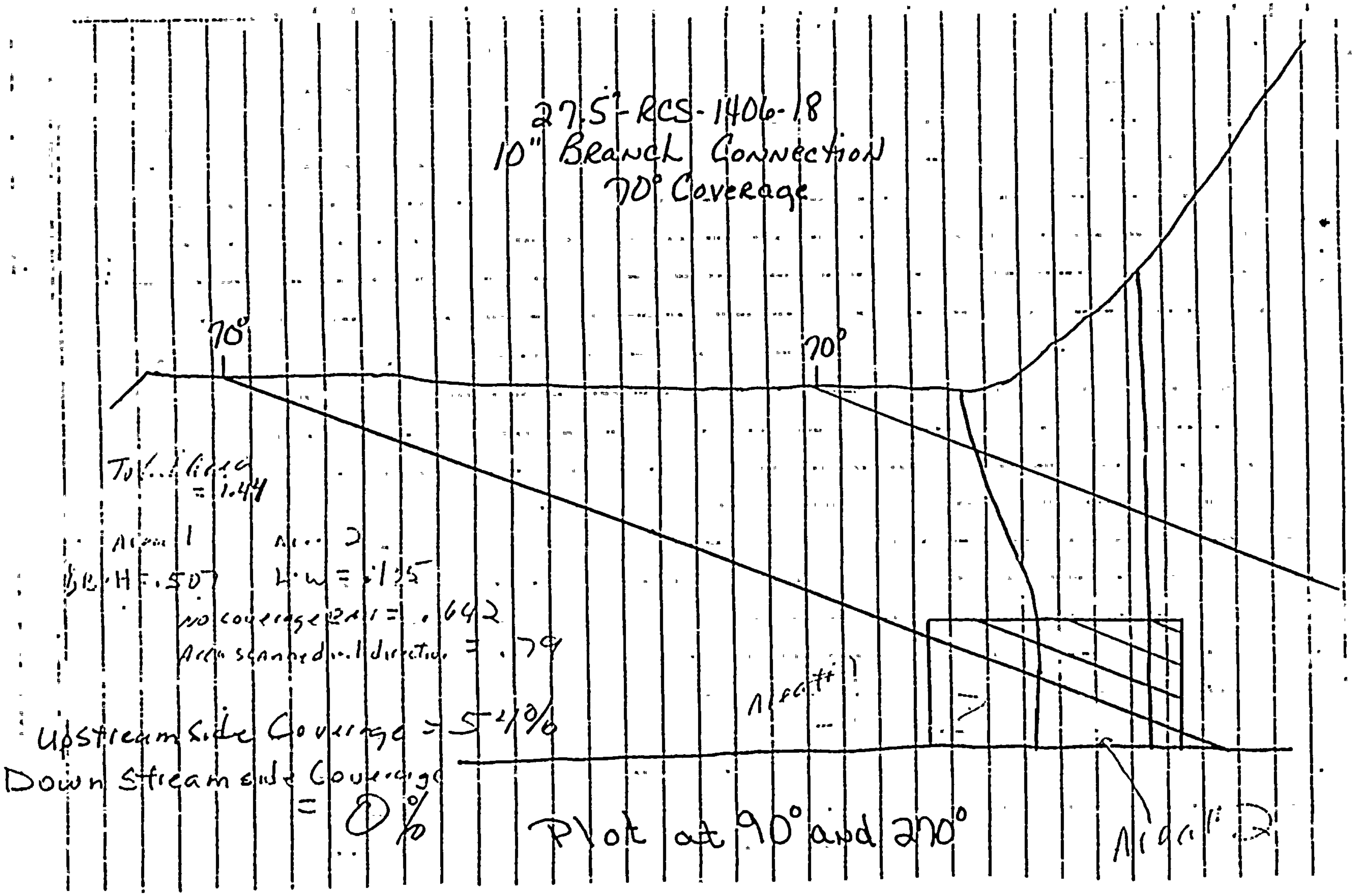
Plot at 90° and 270°

Upstream side Coverage = [illegible]
[illegible]

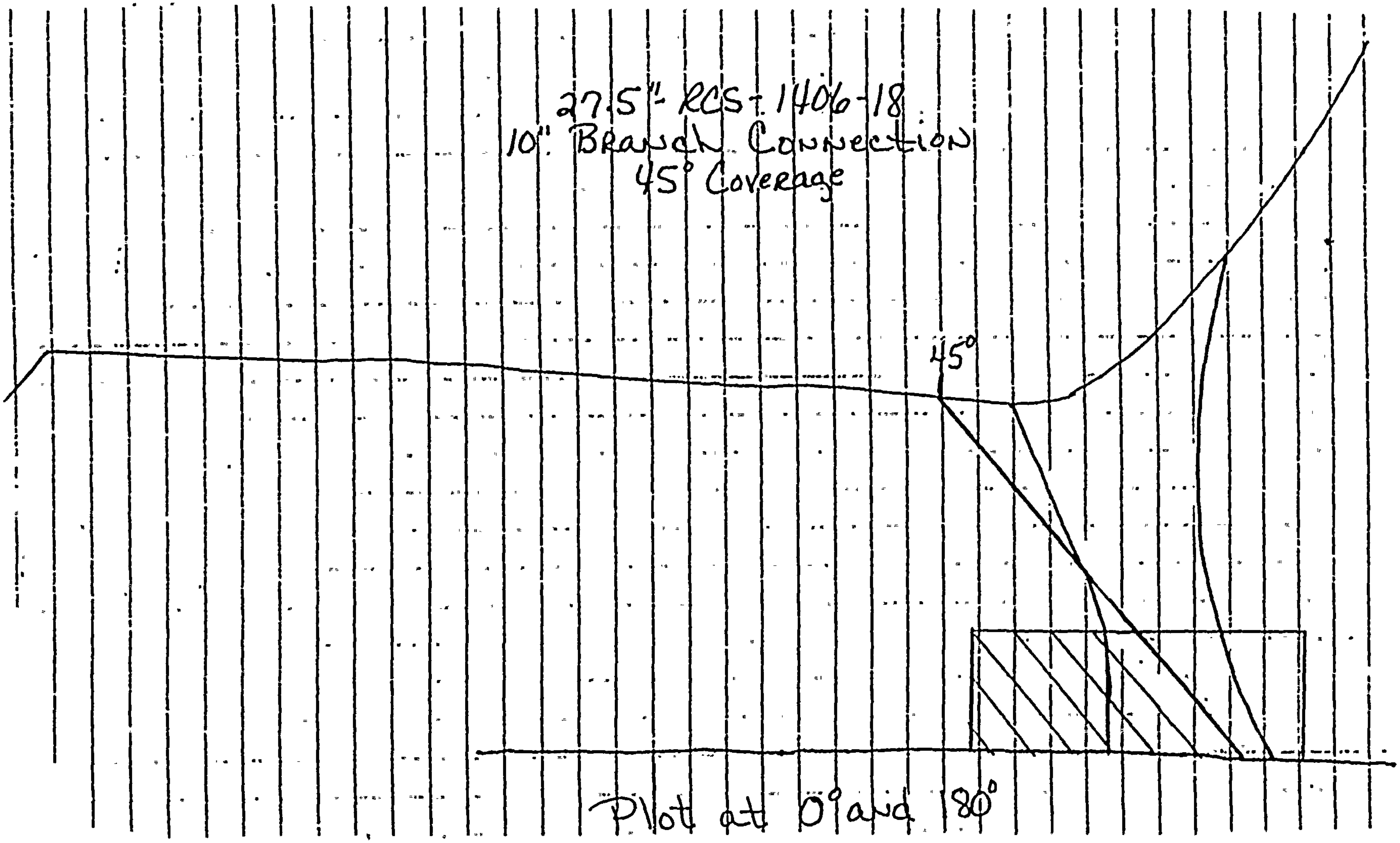
29



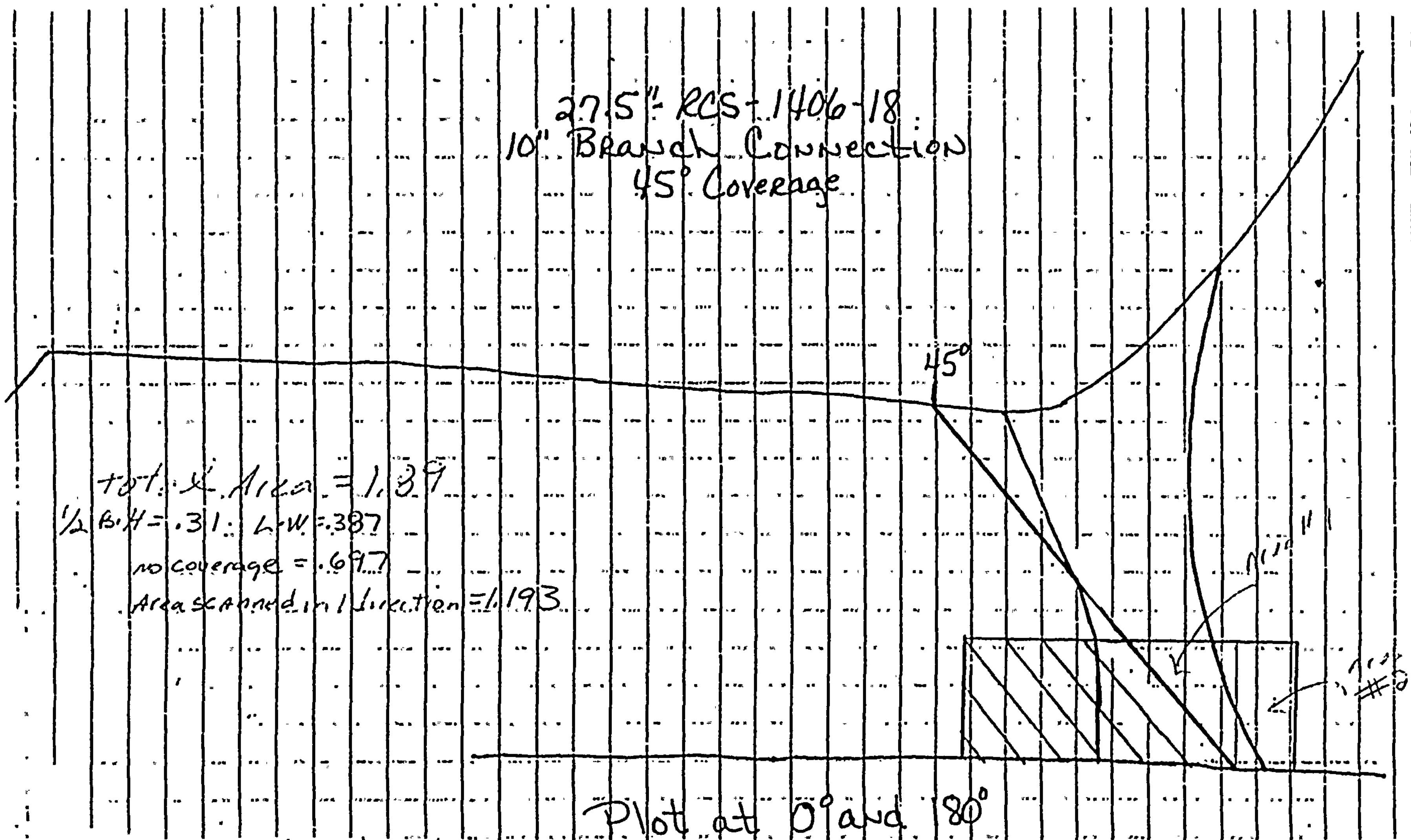

94-4-10 Sheet 3 of 8



27.5"-RCS-1406-18
10" Branch Connection
70° Coverage
70°
70°
= 1.44
Upstream Side Coverage = 54%
Down Stream Side Coverage = 0%
Plot at 90° and 270°



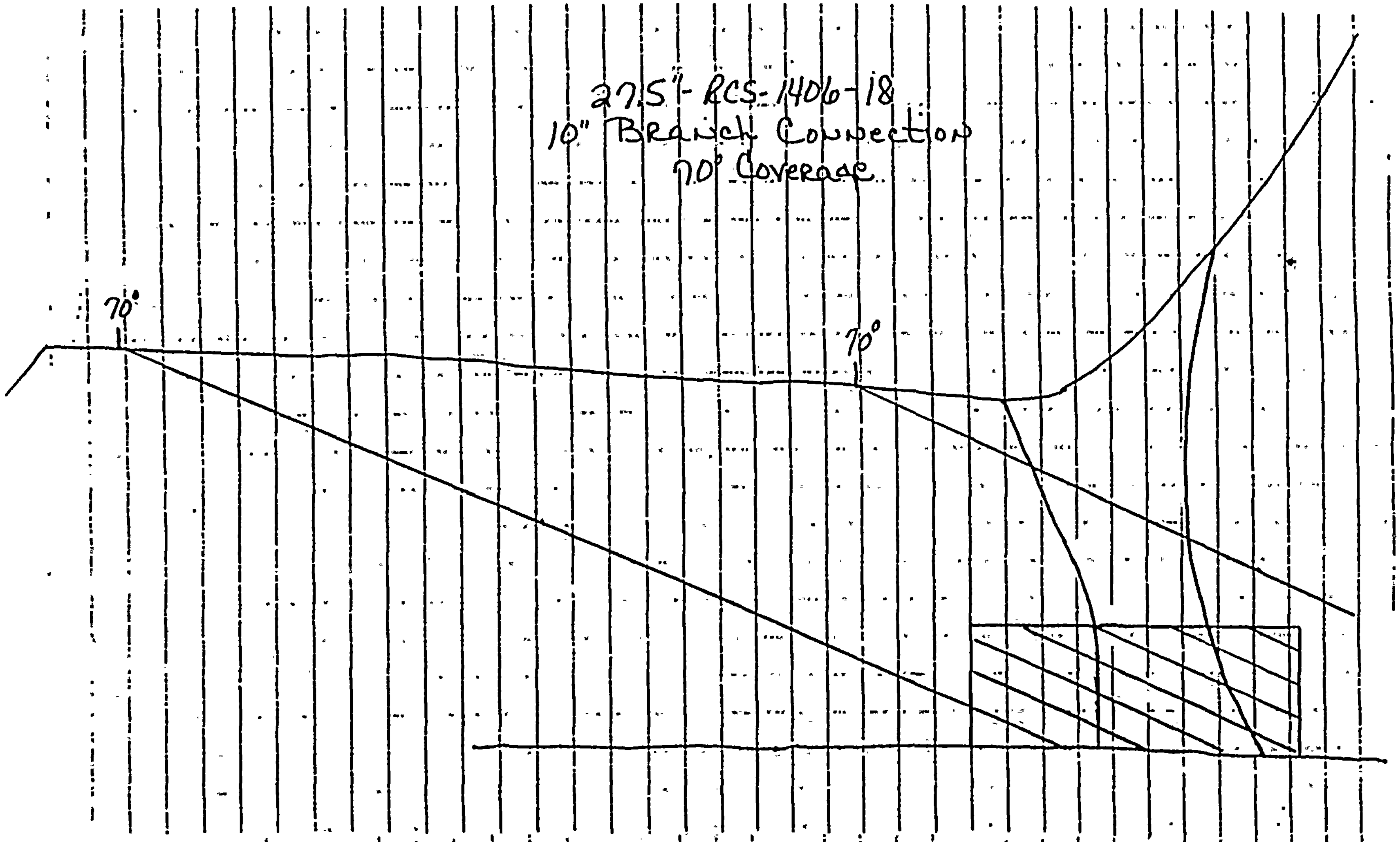

94-4-10 Sheet 5 of 8




Upstream Coverage = 63%

Downstream Coverage = 0%

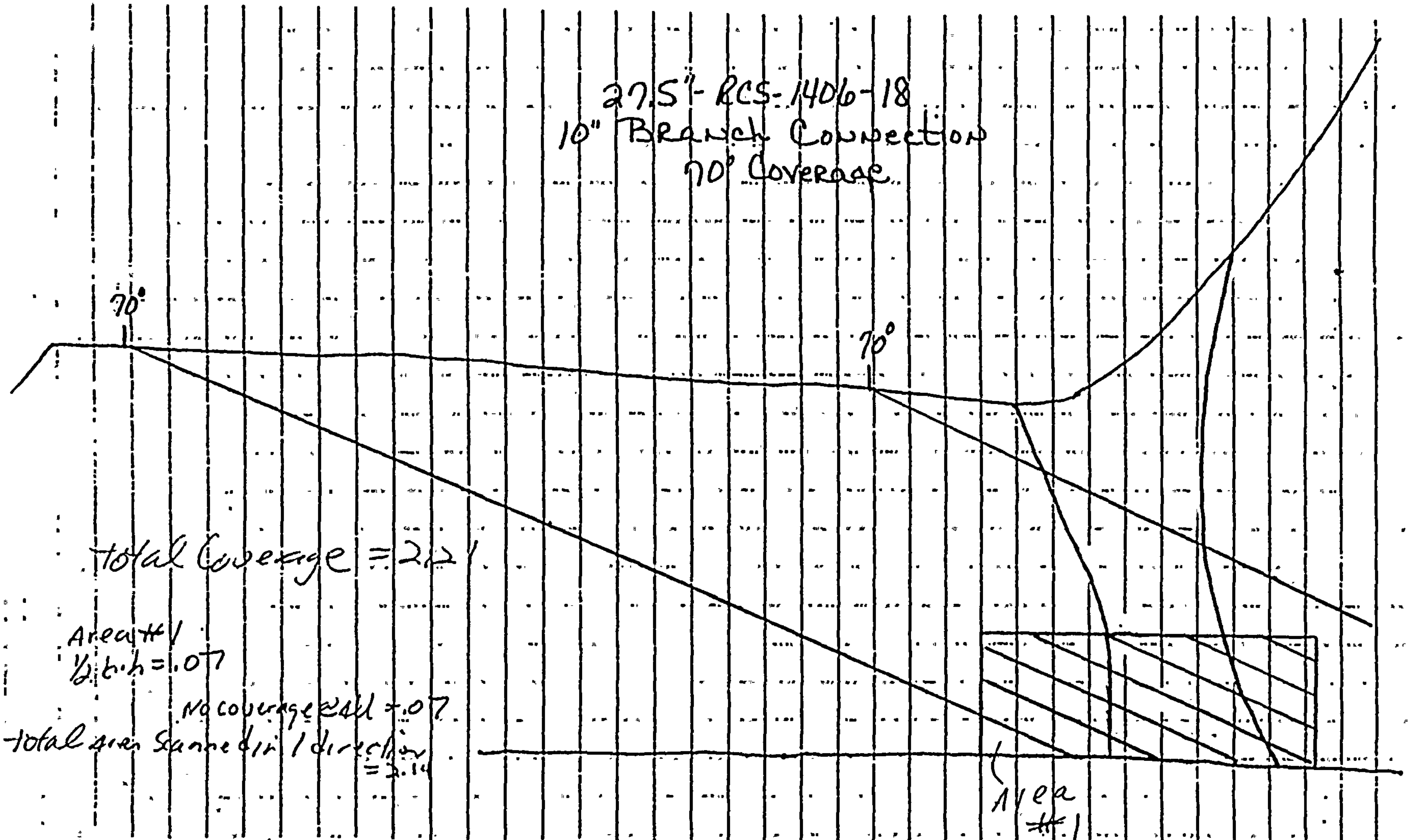
33



Plot at 0° and 180°

94-4-10 Sheet 7 of 8

27.5"-RCS-1406-18
10" BRANCH CONNECTION
70° COVERAGE



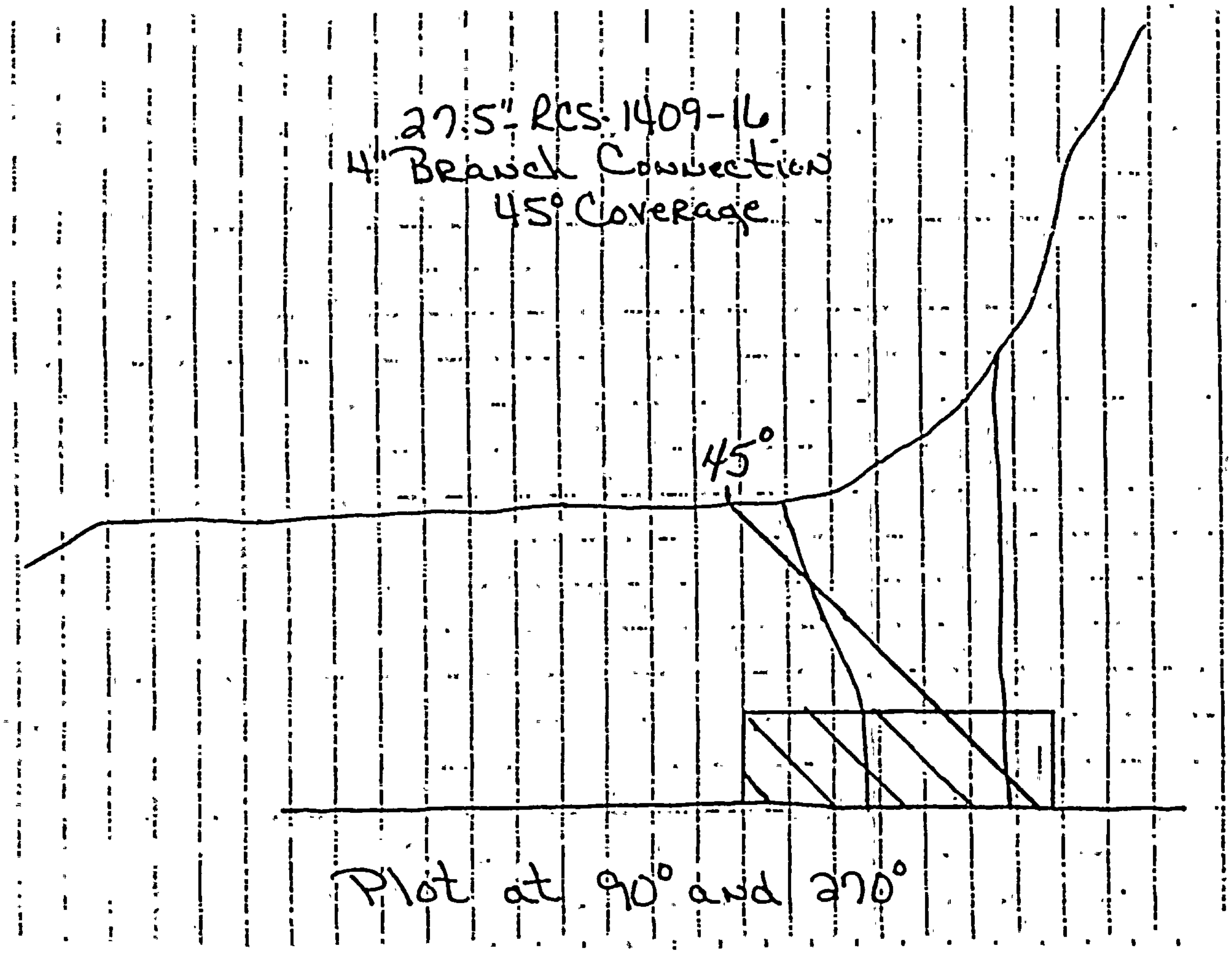

Total Coverage = 2.21

Area #1
½ b·h = .07

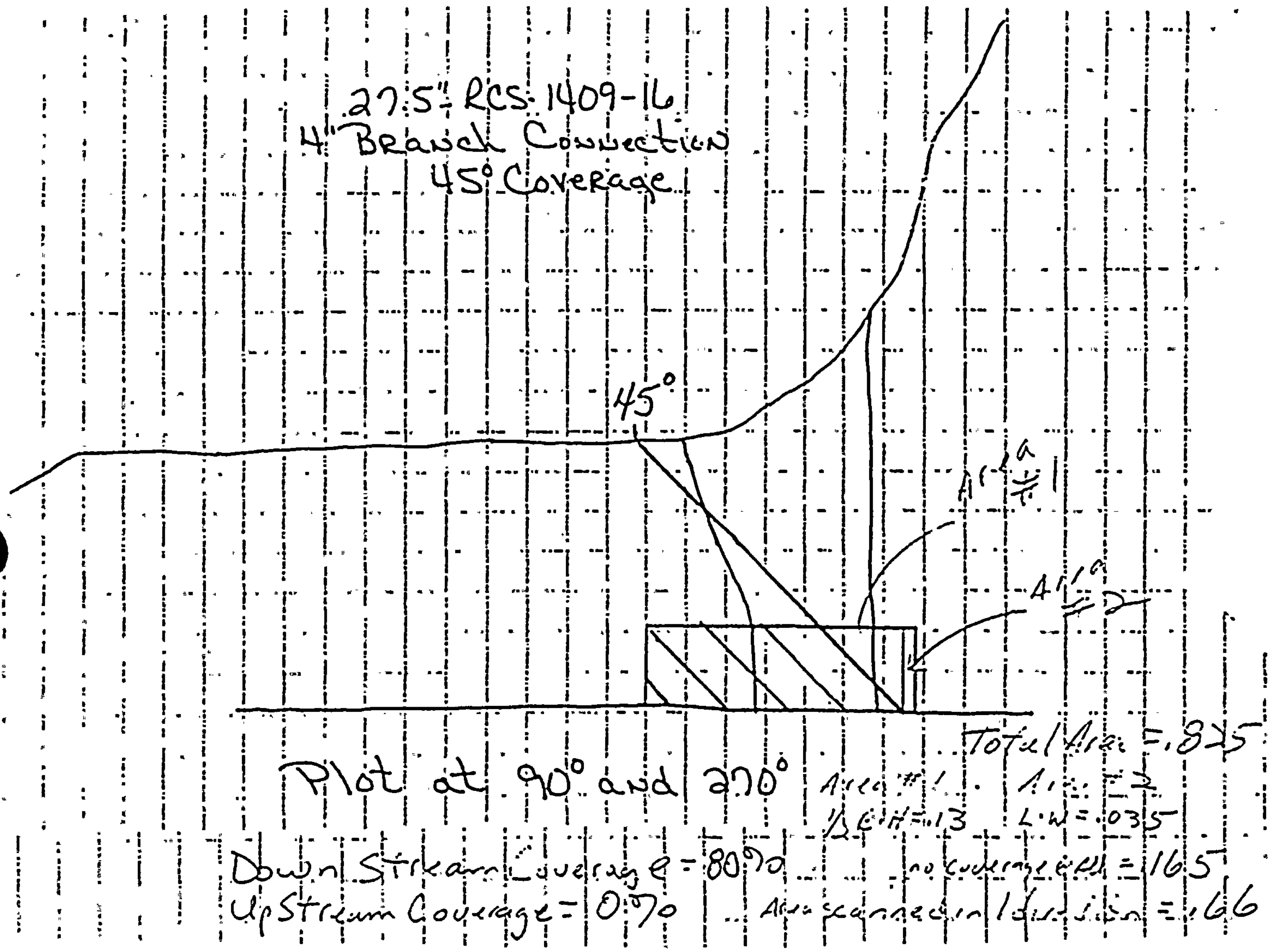
No coverage total = .07
Total area Scanned in 1 direction = 2.14

Upstream Coverage = 97% Plot at 0° and 180°
Downstream Coverage = 0%

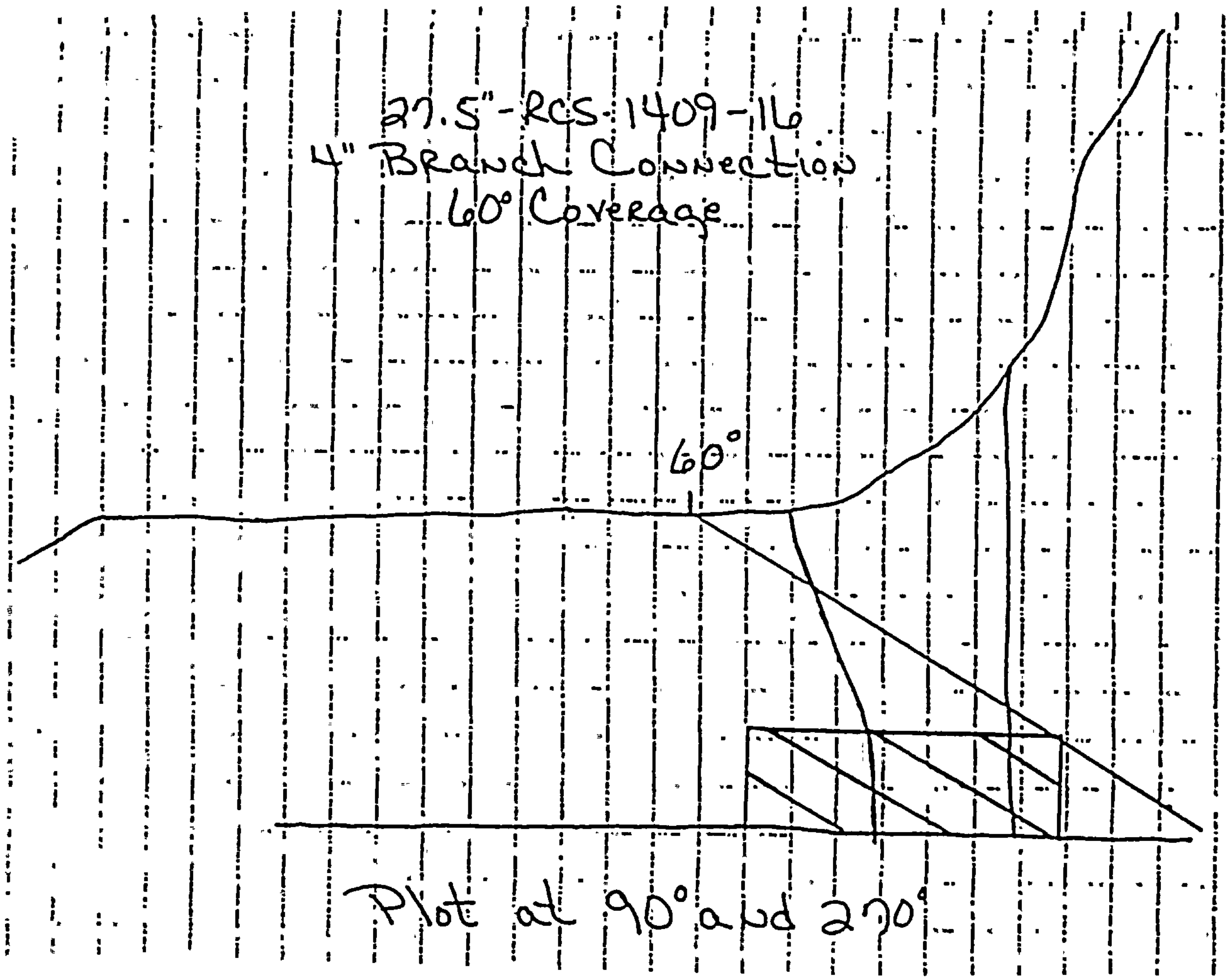
94-4-10 Sheet 8 of 8




94-4-11 Sheet 1 of 8

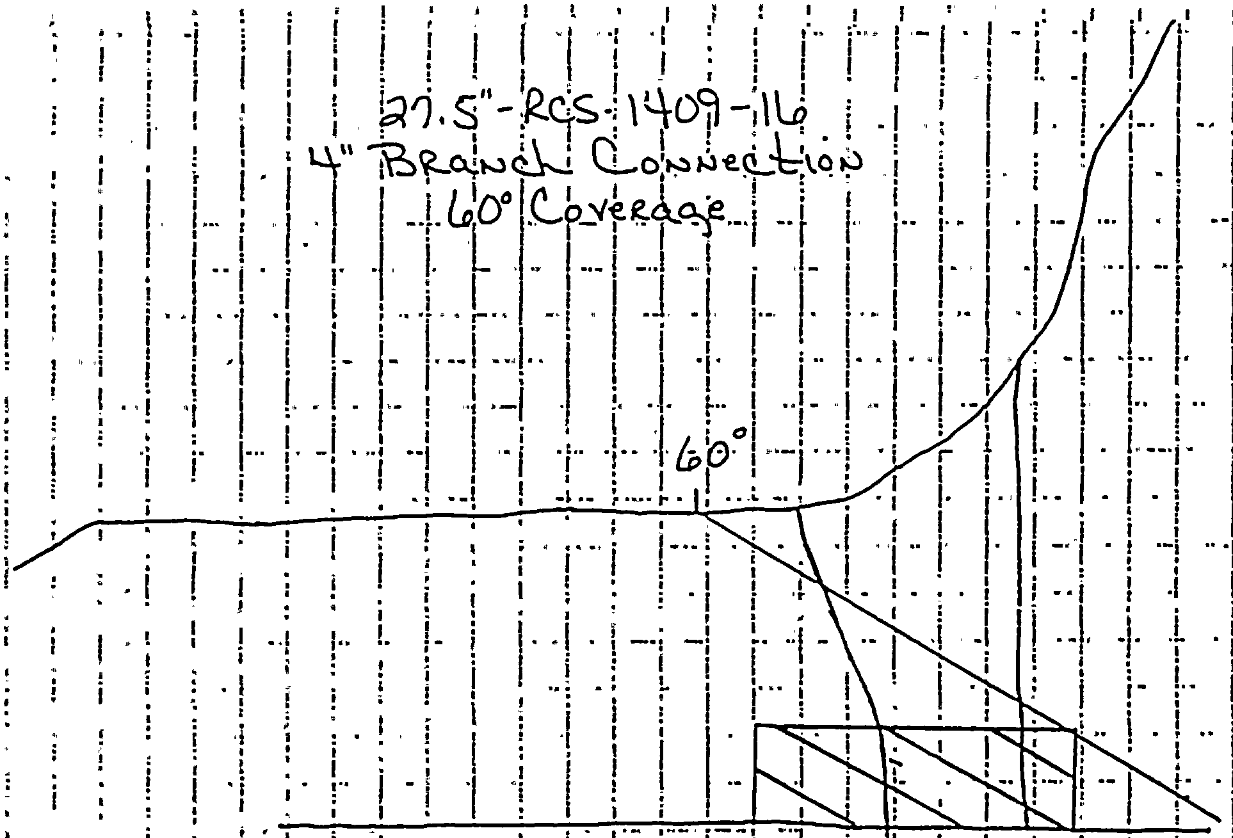



94-4-11 Sheet 2 of 8



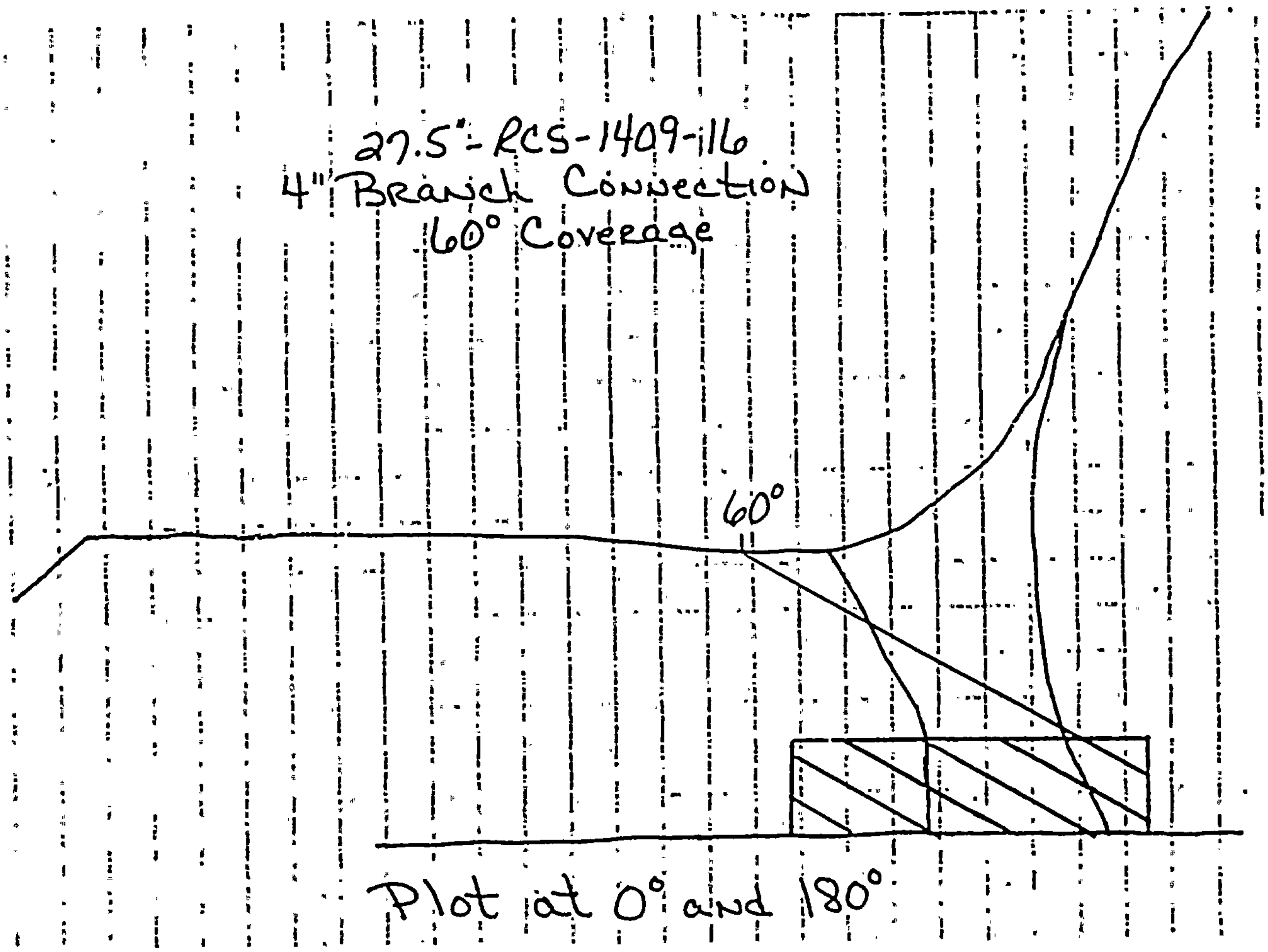

94-4-11 Sheet 3 of 8

27.5"-RCS-1409-16
4" Branch Connection
60° Coverage

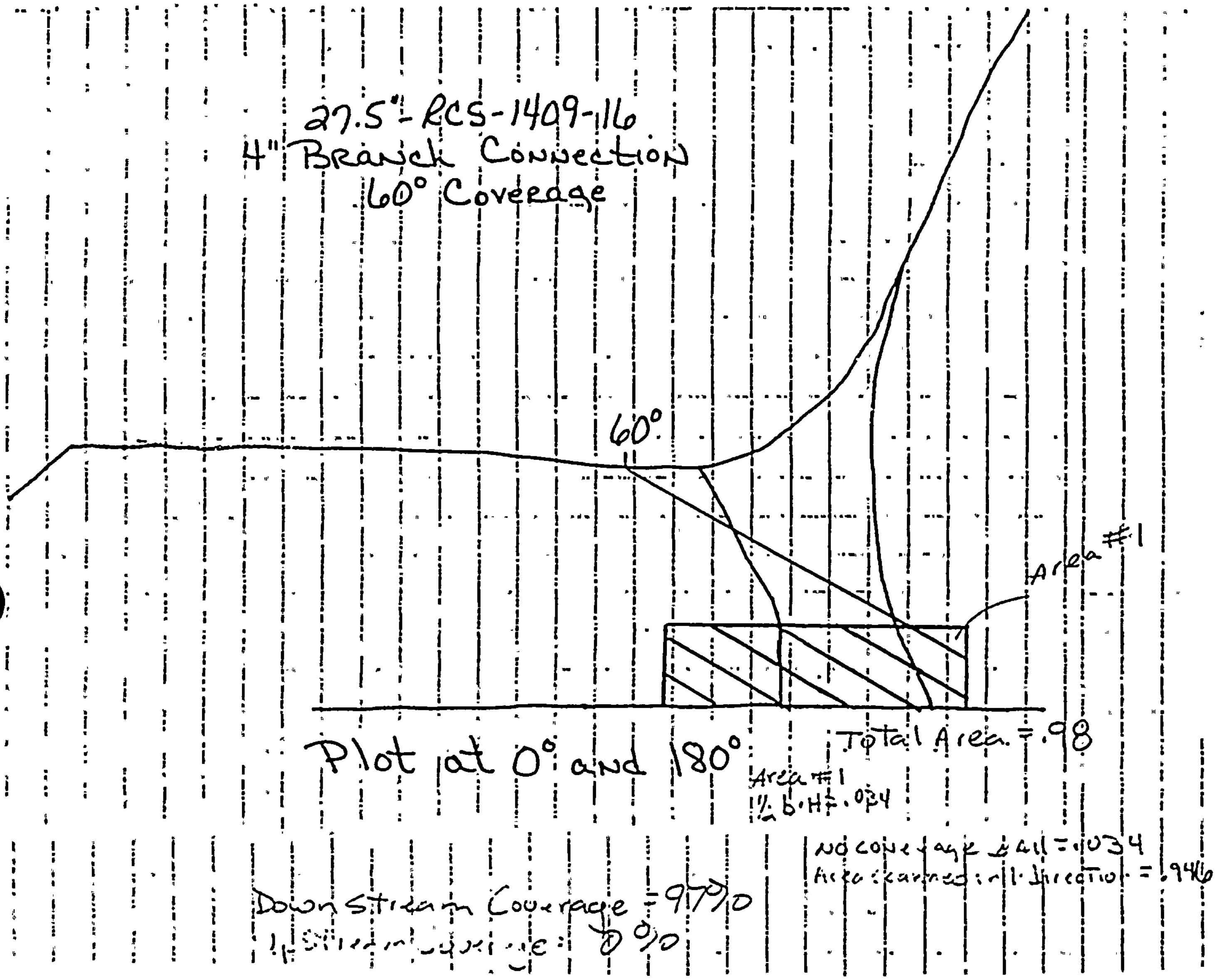



Plot at 90° and 270°

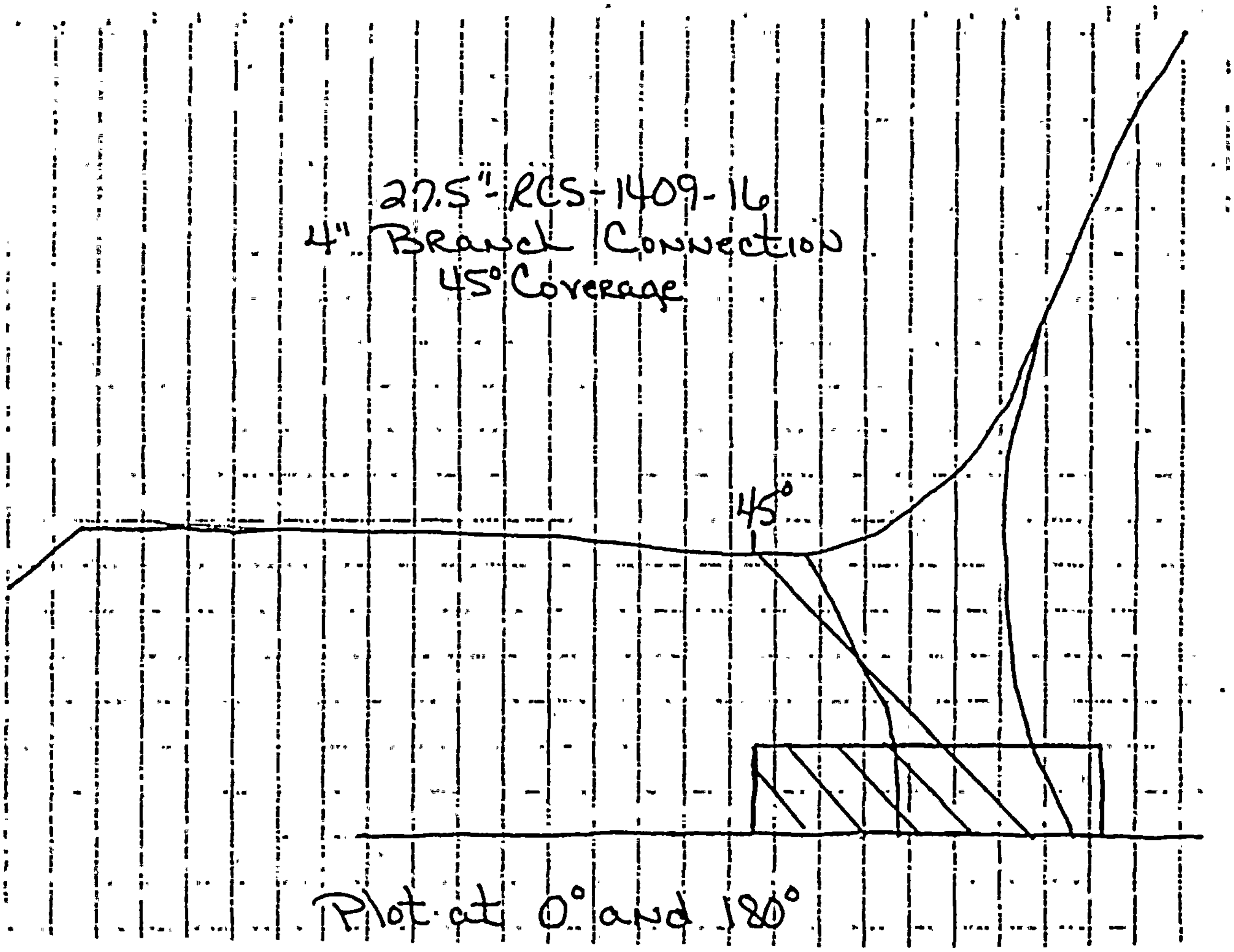
Down Stream Coverage = 100%
Up Stream Coverage = 0%



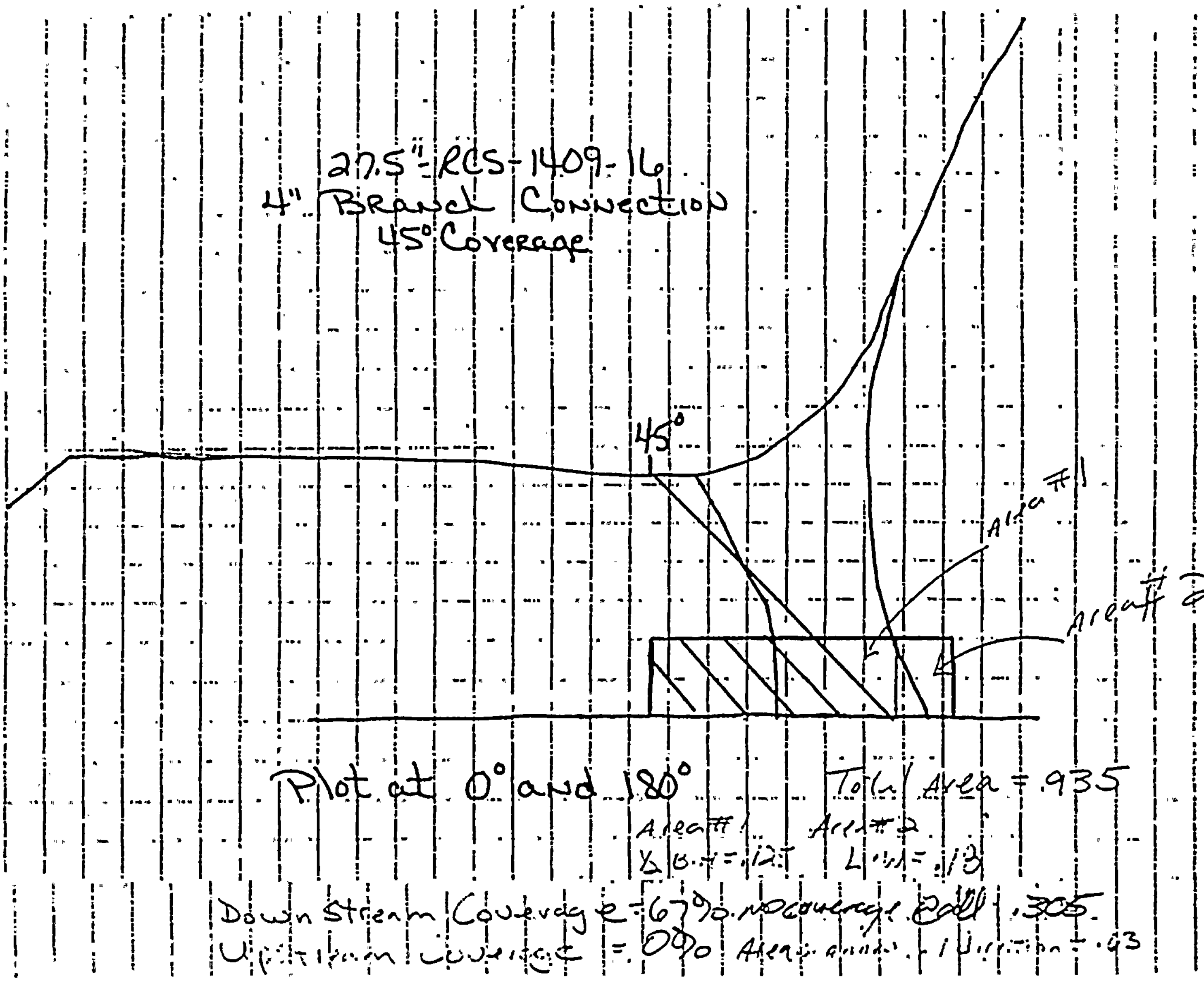

94-4-11 Sheet 5 of 8



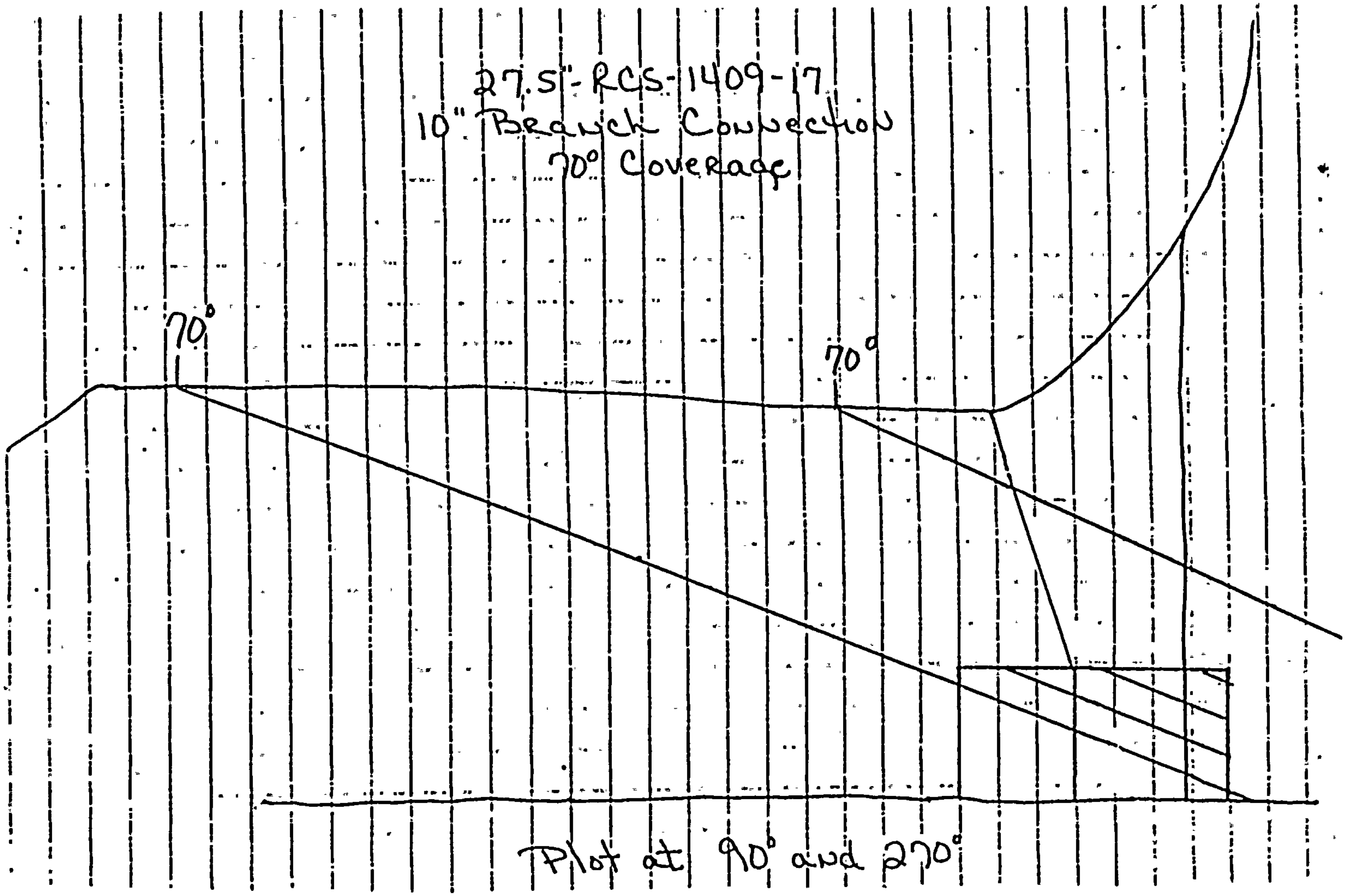

94-4-11 Sheet 6 of 8




94-4-11 Sheet 7 of 8

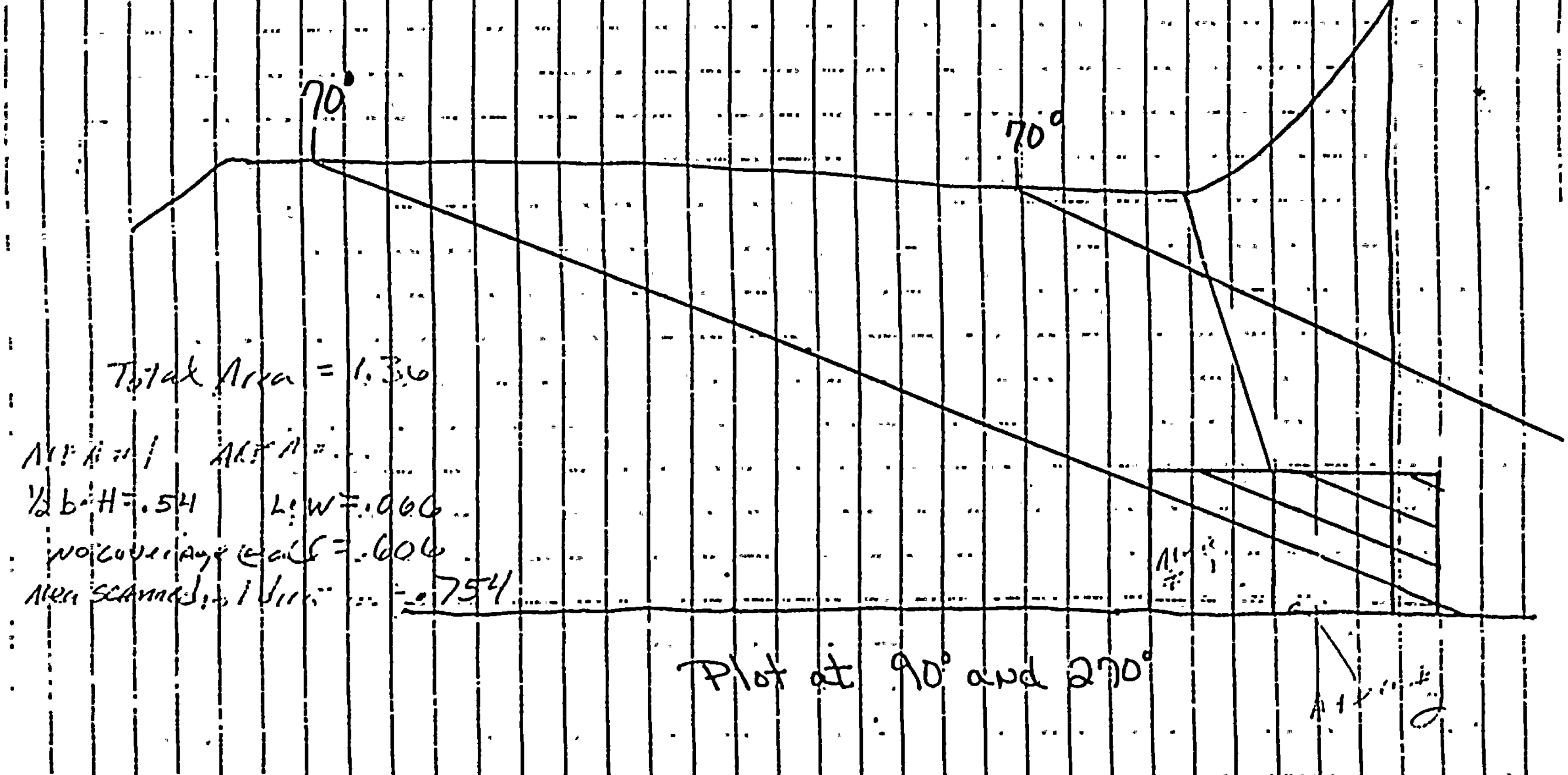



94-4-11 Sheet 8 of 8

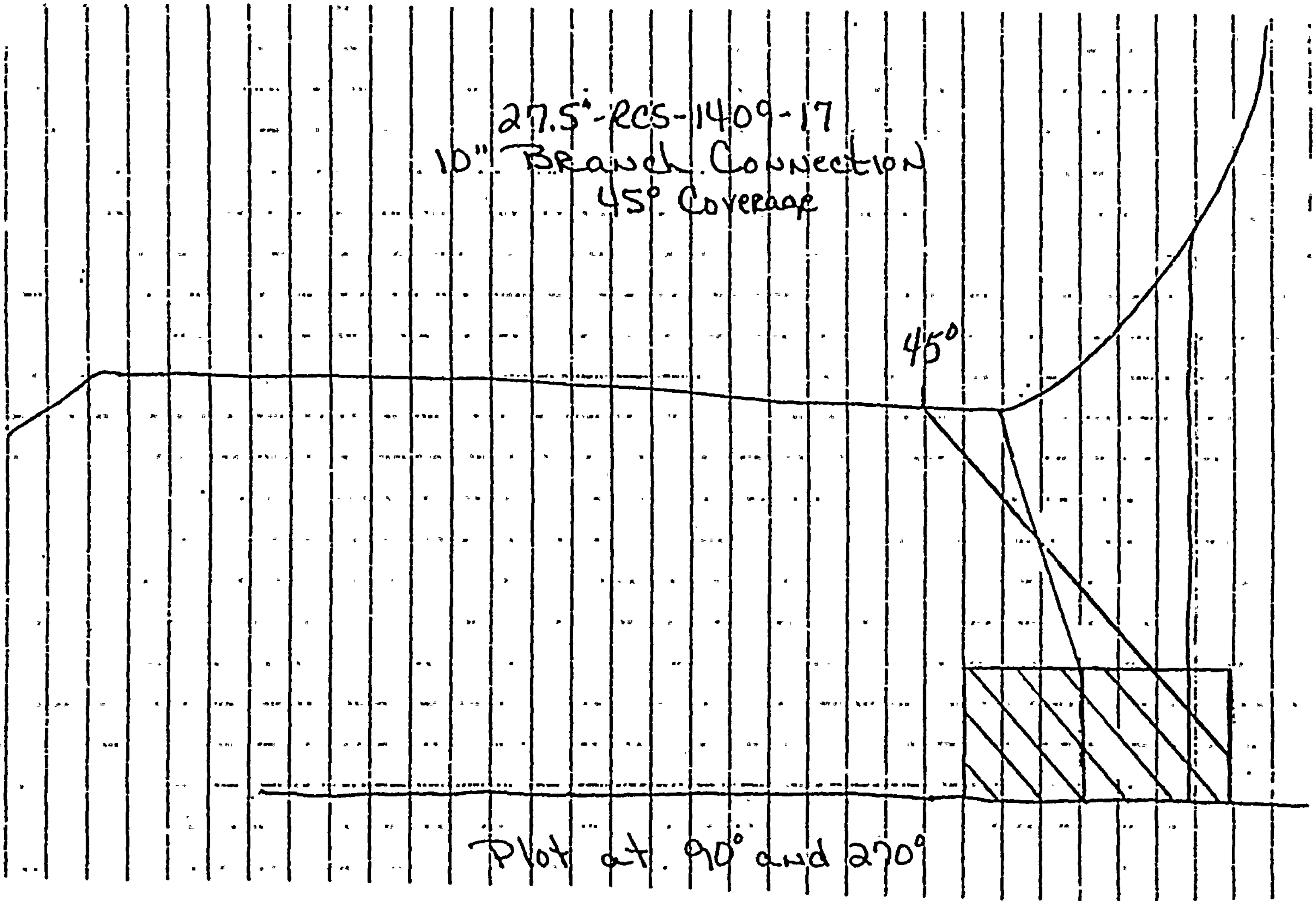



94-4-12 Sheet 1 of 8

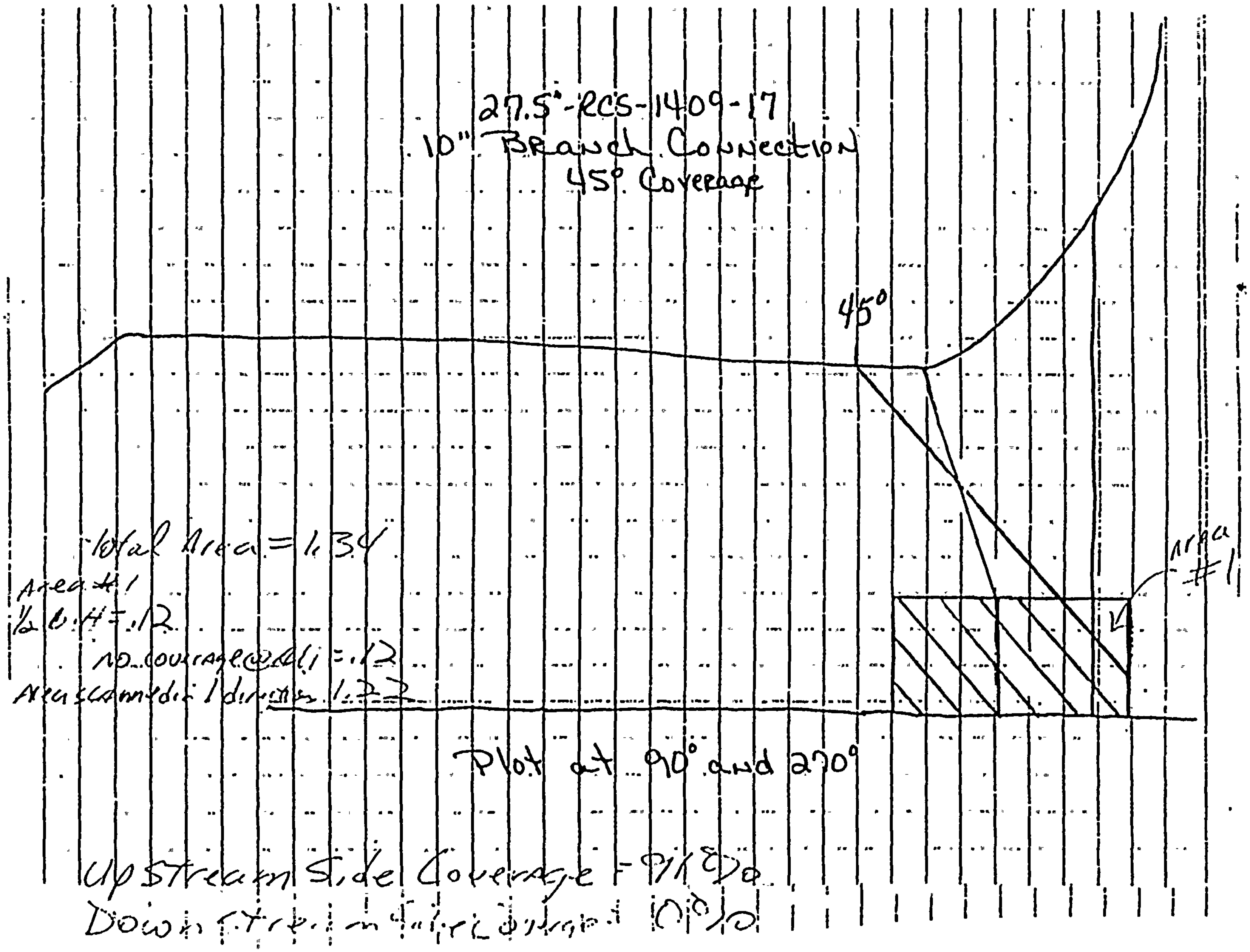
27.5"-RCS-1409-17
10" Branch Connection
70° Coverage

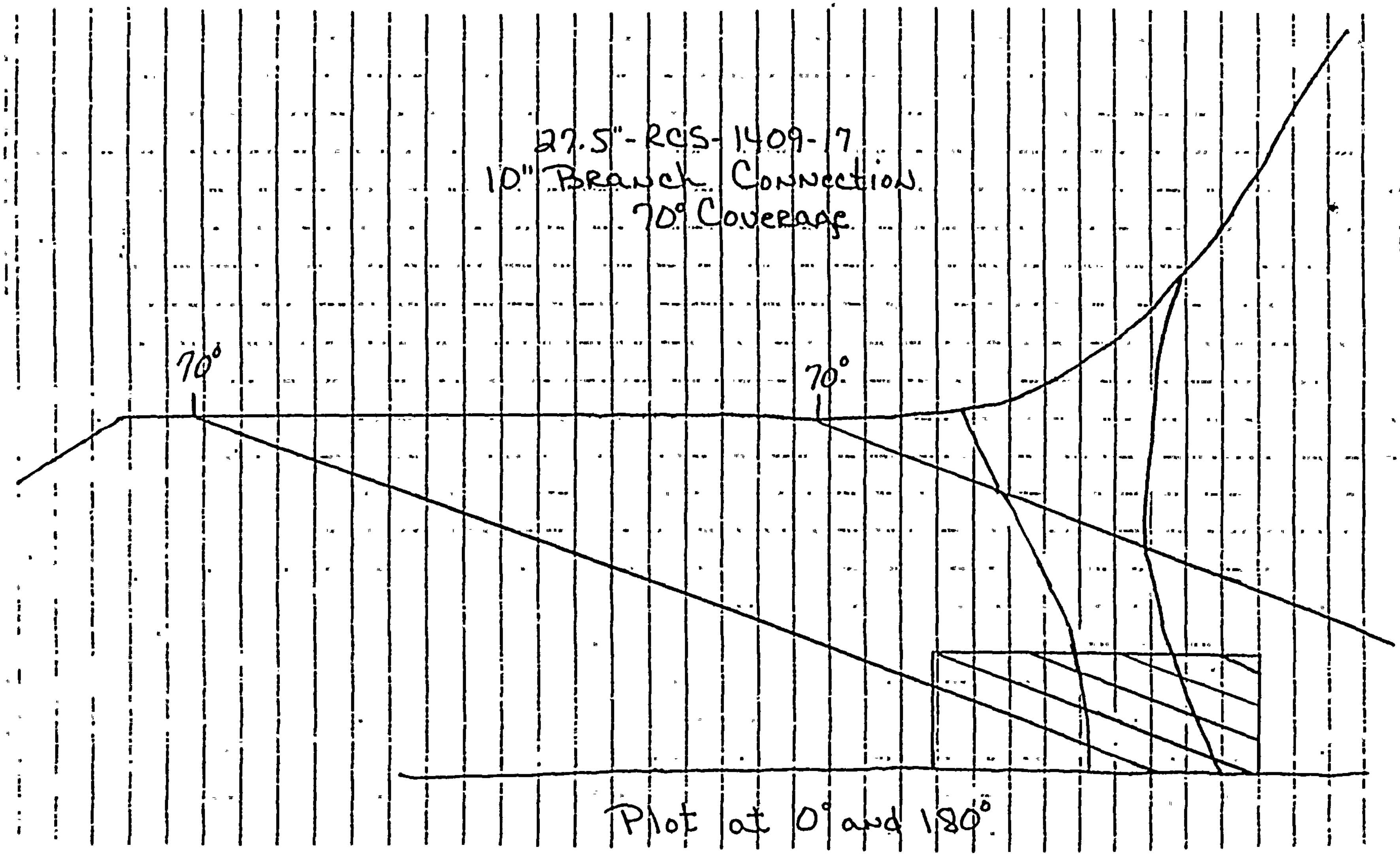



Up Stream Side Coverage = 55%
Downstream Side Coverage = 0%

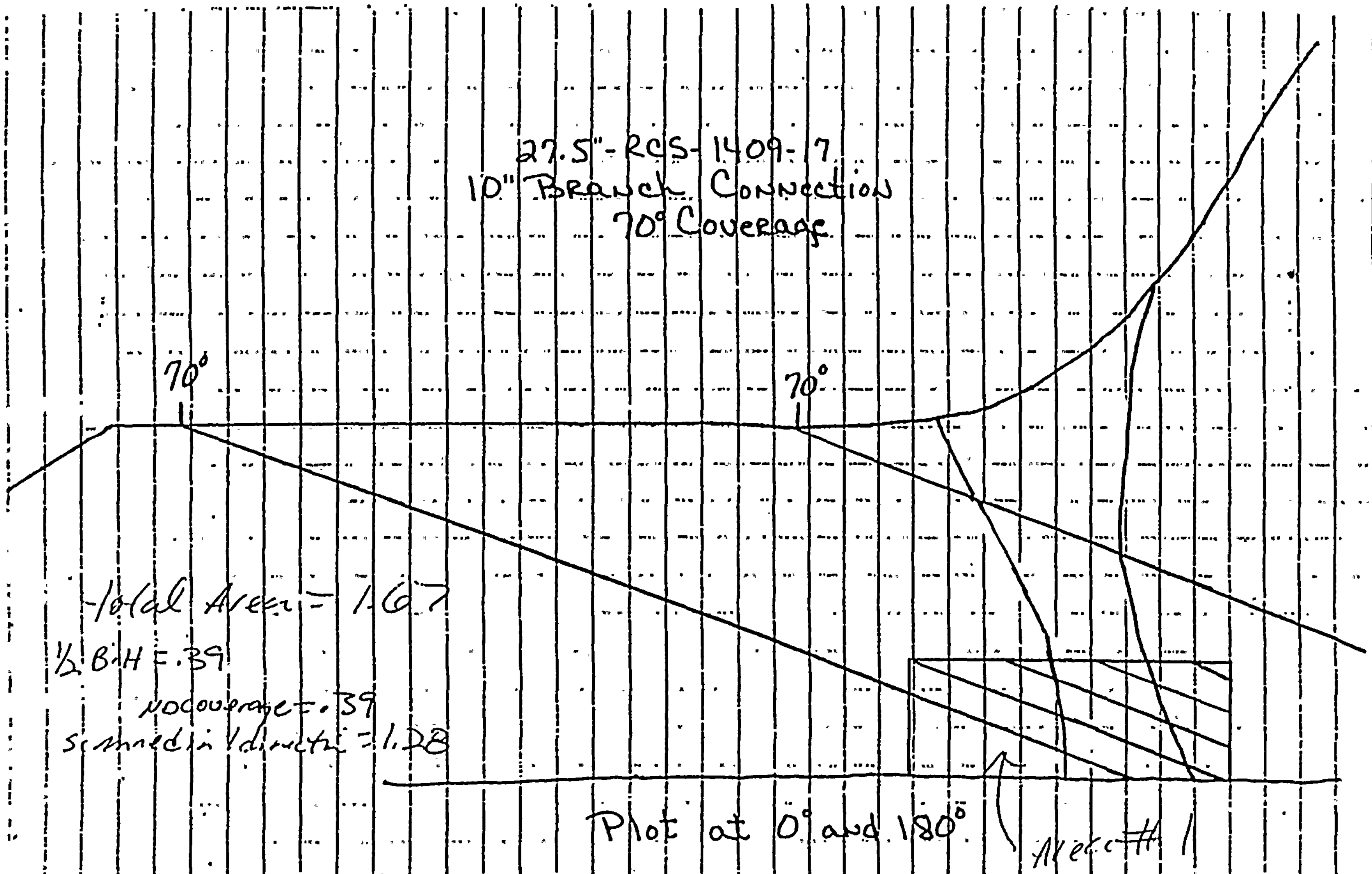



94-4-12 Sheet 3 of 8


27.5"-RCS-1409-17
10" Branch Connection
45° Coverage
45°
Total Area = 1.34
Area #1
½ B·H = .12
Area #1
Plot at 90° and 270°
Upstream Side Coverage = 91%

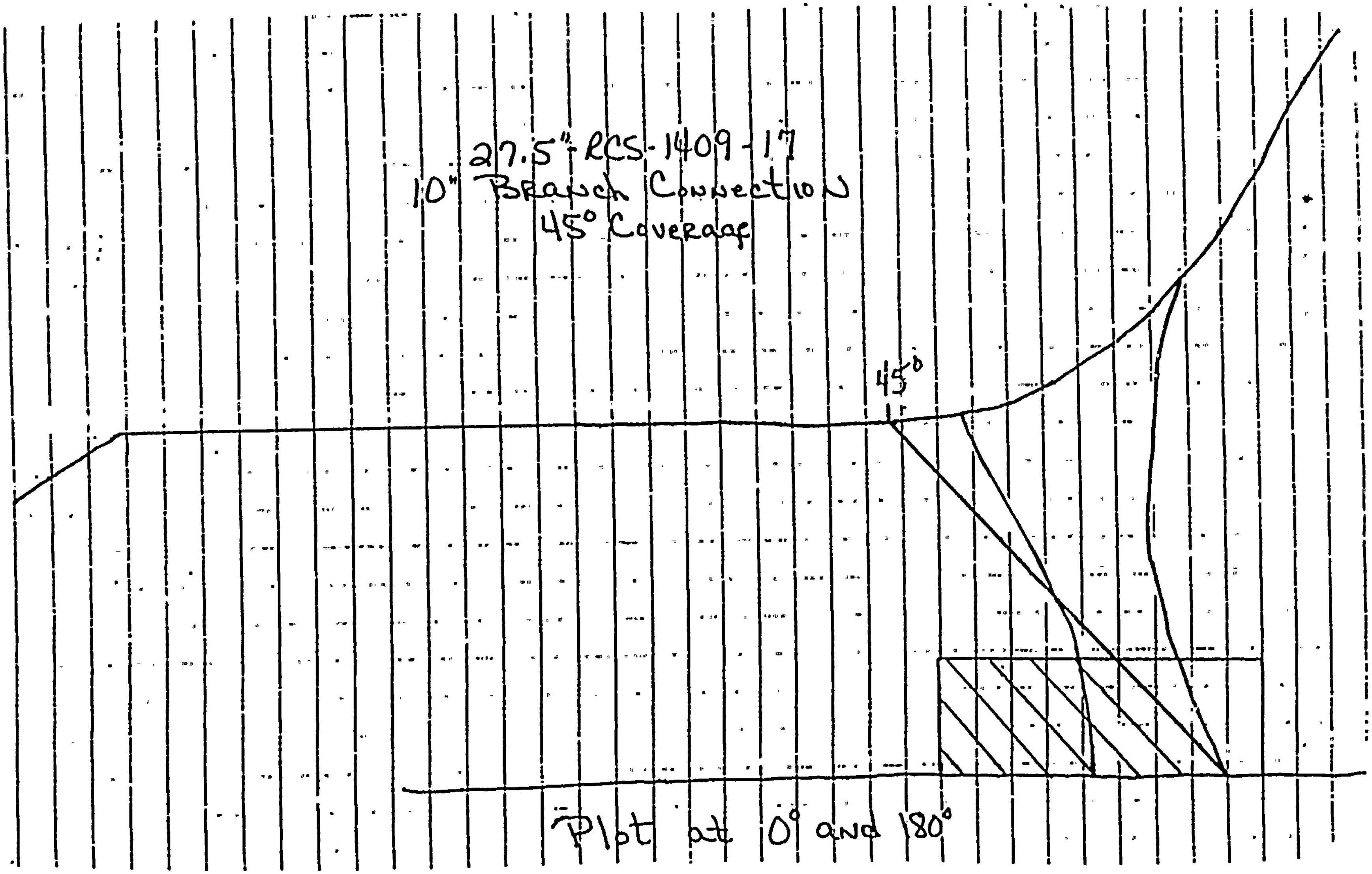


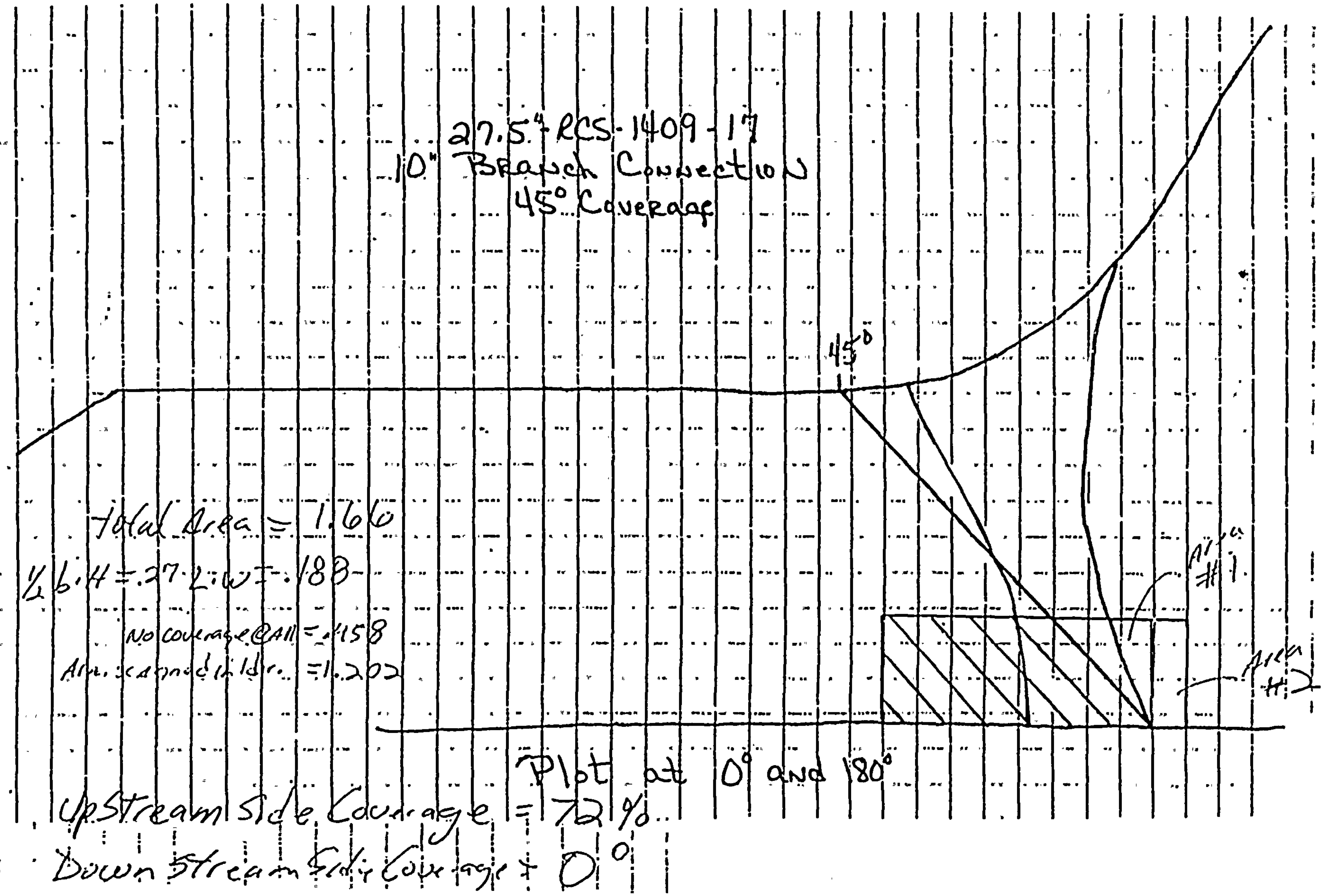

94-4-12 Sheet 5 of 8




Total Area = 1.67
½ B·H = .39
no coverage = .39
scanned in 1 direction = 1.28

Upstream Side Coverage = 77%
Down Stream Side Coverage = 0%

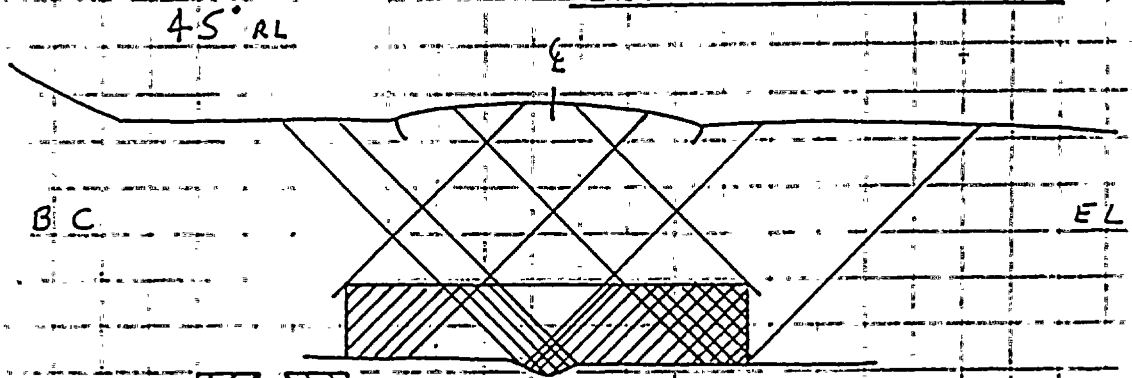

27.5"-RCS-1409-17
10" Branch Connection
45° Coverage
45°
Plot at 0° and 180°


27.5" RCS-1409-17
10" Branch Connection
45° Coverage
45°
Total Area = 1.66
½ b·H = .27 L·W = .188
No coverage @ All = .458
Area scanned = 1.202
Area #1
Area #2
Plot at 0° and 180°
UpStream Side Coverage = 72%
Down Stream Side Coverage = 0%

## 14" - RHR - 1401-1     C. R. V. SKETCH

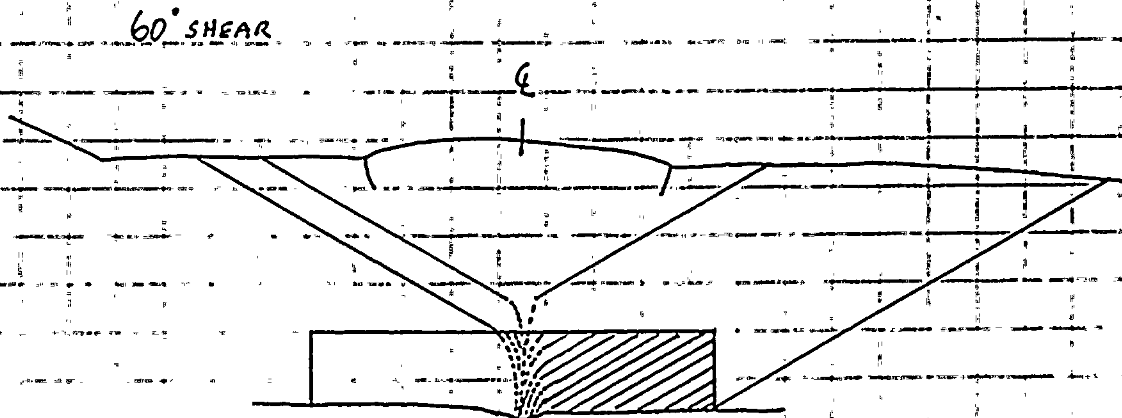
FLOW

CRV AREA = 0.945 in²




COVERAGE ONE DIRECTION ONLY = 0.54 in² = 57% CRV

NO COVERAGE = 0.16 in² = 17% CRV



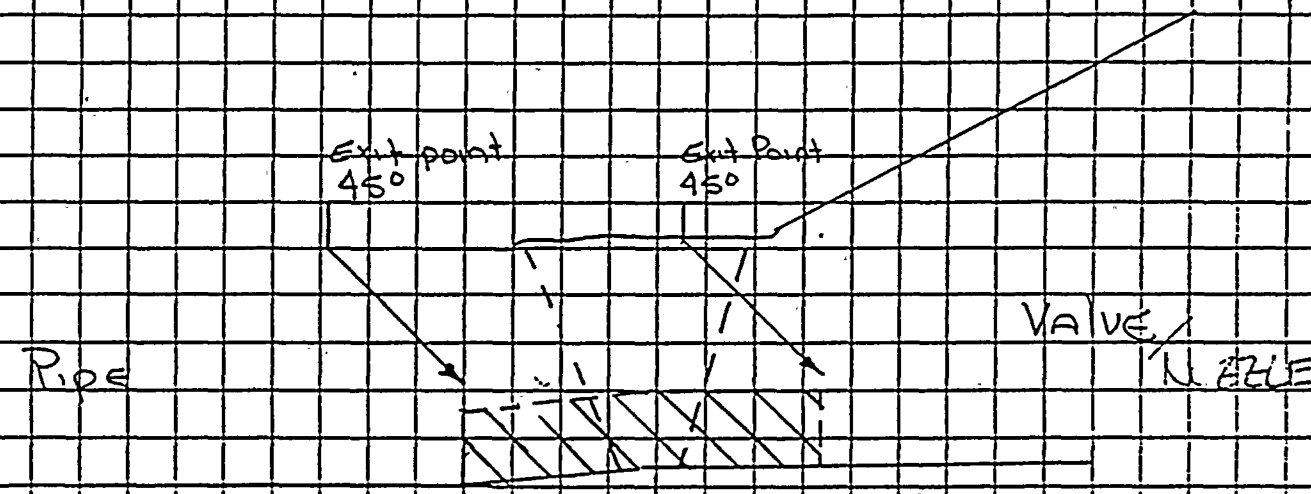

70% CRV achieved from elbow side.
35% CRV achieved from B.C. side.

Pipe to Valve Configuration

14"-140

14"-RHR-1401-5

45° 1/2 VEE Examination Technique Used
60° 1/2 VEE Examination Technique Used




100% Coverage is claimed in one Direction

0% Coverage in the 2nd direction

94-4-14

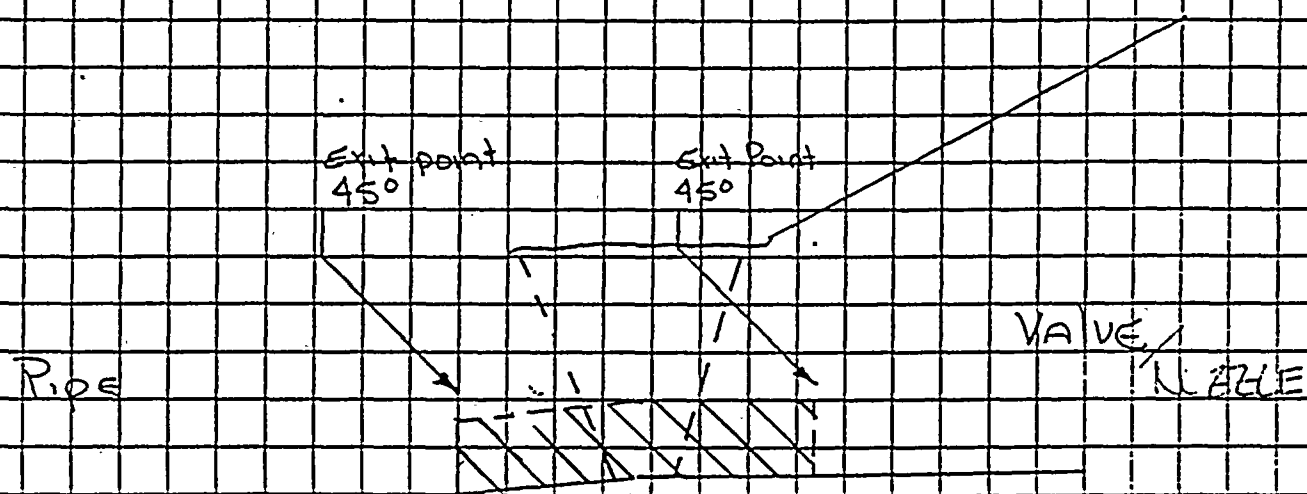
Pipe to Valve Configuration

14"-140

14"-RHR-1401-6

45° 1/2 VEE Examination Technique Used

60° 1/2 VEE Examination Technique Used




100% Coverage is claimed in one Direction

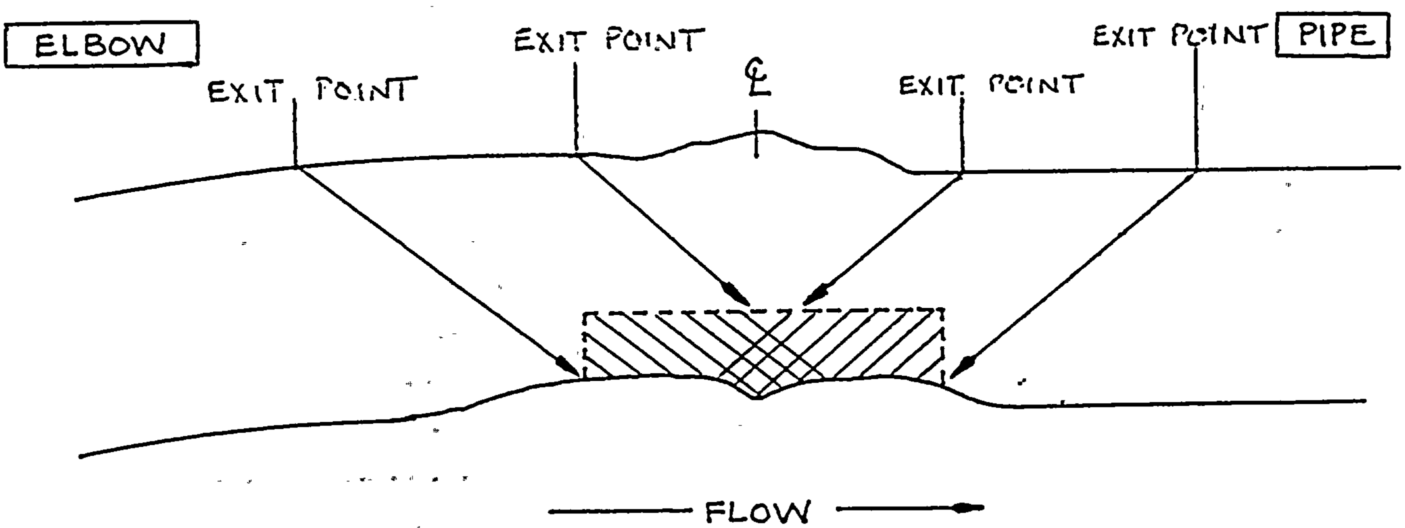
0% Coverage in the 2nd direction

94-4-15

RHR
FROM RCS LOOP A
HOT LEG

45° CRV
1/2 V

14"-RHR-1401-9



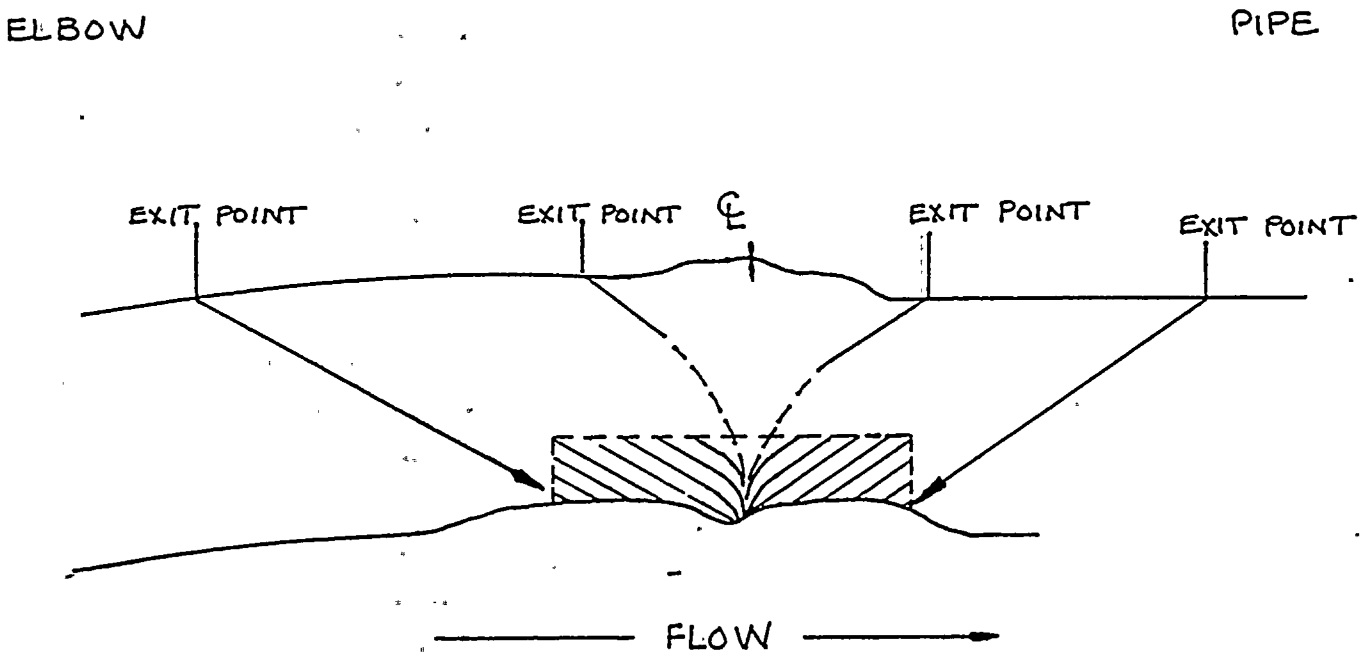

CRV
53% from Pipe Side
54% from Elbow Side

94-4-16 Sheet 1 of 2

RHR
FROM RCS LOOP A
HOT LEG

60° CRV.
1/2 V

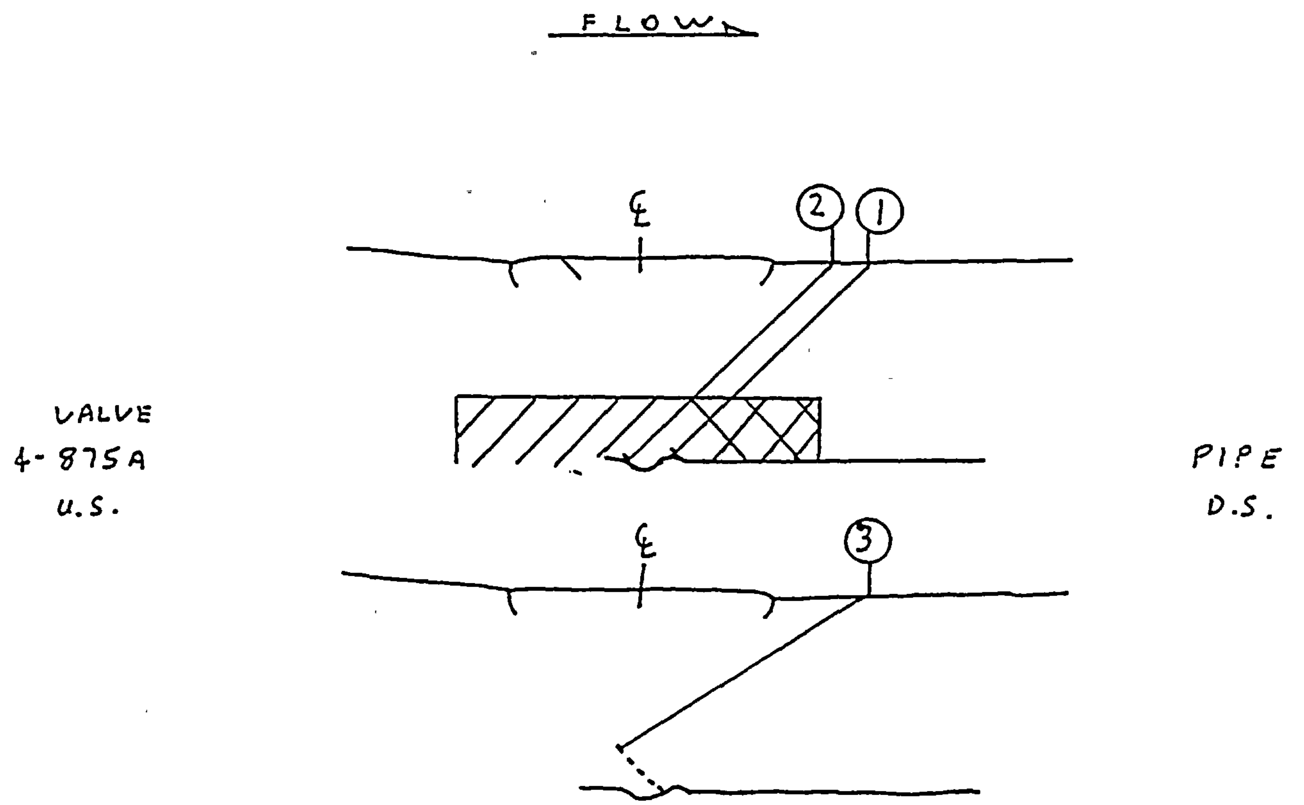
14"-RHR-1401-9




94-4-16 Sheet 2 of 2

10"-SI-1401-14

45° ½ VEE & 60° ½ VEE




① 45° COUNTERBORE SEEN INTERMITTENT 360° AT VARIOUS AMPLITUDES

② 45° ROOT GEOMETRY SEEN INTERMITTENT 360° AT VARIOUS AMPLITUDES.

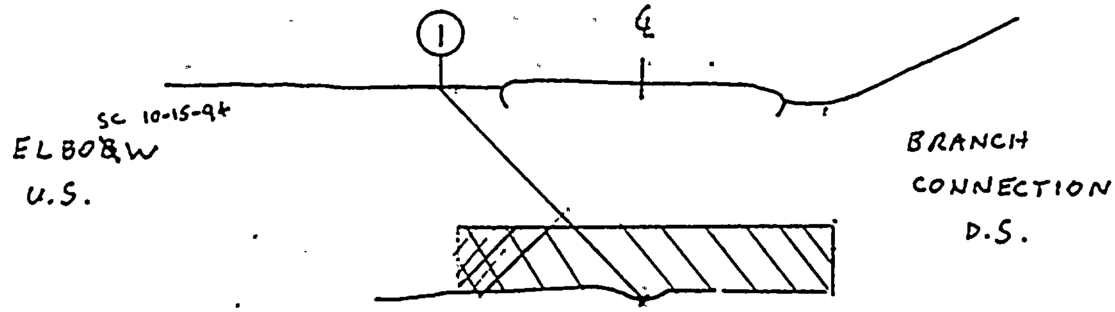
③ 60° ROOT GEOMETRY - VELOCITY/ANGLE CHANGE CAUSED BY GRAIN STRUCTURE. SEEN INTERMITTENT 360° AT VARIOUS AMPLITUDES.

100% CRV from Pipe Side
20% CRV from Valve Side

94-4-17

10"-SI-1401-18

FLOW




① 45° ROOT GEOMETRY SEEN INTERMITTENT 360° AT VARIOUS AMPLITUDES.

ONE SIDED EXAM

100% CRV from elbow side
18% CRV from BC side

94-4-18