REPORT ON INSTRUMENTATION PORT

COLUMN ASSEMBLY LEAKAGE

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT UNIT 4

APRIL 27, 1987

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1.0 EXECUTIVE SUMMARY

In March, 1987 during an inspection of the Turkey Point Unit 4 Reactor Pressure Vessel (RPV) head area, Florida Power & Light (FPL) found that a significant amount of boric acid crystals had been deposited on the reactor vessel head due to a leak in the lower seal (conoseal) of a thermocouple instrumentation port column assembly. The purpose of this report is to identify the components which have been affected by the conoseal leakage, to identify the potential leakage mechanisms, to describe the actions FPL has taken for these components, and to describe the actions FPL has taken, or is planning to take, to prevent recurrence of this type of event.

The lower conoseal joint consists of a stainless steel conoseal gasket between stainless steel male and female flanges. The flanges are held in place by use of a carbon-steel clamp. This design has been used at Turkey Point since it first began operation and has also been used on other Westinghouse nuclear plants.

The leak in the conoseal was first identified in August, 1986 as a result of a pre-critical containment walkdown by maintenance personnel. The maintenance superintendent subsequently described the leak as a "wisp of steam". Engineering evaluated the leak and determined that operation with the leak would be acceptable based upon factors such as the small amount of the leak, a low potential for an increase in the leak, the predicted rate of corrosion of the clamp, daily monitoring of the reactor coolant leak rate during operation, and performance of another inspection of the conoseal leak within six months. Accordingly, Unit 4 was restarted in August, 1986.

In October, 1986, another inspection of the conoseal leak was performed during a short outage. The leak rate did not appear to be greater than observed in August, and no significant corrosion or pitting of the clamp was found. However, a relatively small amount of boric acid crystals was found at the conoseal and adjacent areas. Based upon the results of this inspection and the August, 1986 evaluation, FPL Engineering determined in February, 1987 that Unit 4 could be operated until April, 1987 without another inspection of the conoseal leakage. During another outage, on March 13, 1987, FPL learned that the actual corrosion rates may be greater than those used in the August 1986 evaluation. FPL brought the unit to cold shutdown and performed another inspection of the conoseal leak and found a significant amount of boric acid crystals on the reactor vessel head area.

After discovery of the boric acid crystals on the reactor head area in March 1987, FPL performed extensive inspections to identify the extent of the items which were in contact with boric acid deposits. These included inspections of items in the area of the reactor vessel head, walkdowns and analysis of equipment in containment which have been environmentally qualified under 10CFR50.49, and a more general walkdown of equipment in the containment to identify any other items which may have been affected by the conoseal leakage.

Following these walkdowns and inspections, FPL took several actions for those items which had evidence of boric acid deposition. In general, these actions consisted of noting the conditions of items; cleaning the items which had boric acid deposits; performing visual inspections and non-destructive examinations (NDE), as appropriate for the cleaned items; evaluating the results of the inspections and NDE; and repairing or replacing items as warranted. Specifically, the equipment addressed during these walkdowns and inspections were as follows:

- Reactor Vessel Head Dome, Flange and Penetrations
- Reactor Vessel Head Studs, Nuts and Washers
- Reactor Vessel Flange, Flange Side and Stud Holes
- Annulus Region, Reactor Vessel Shell, Insulation, Nozzles and Nozzle Supports
- Thermocouple Column Assembly and Conoseal
- Control Rod Drive Mechanisms (CRDM) and Rod Position Indicators (RPI), Electrical Connectors, Cables and Instrument/Control Equipment in Locality of Leak
- CRDM Coolers
- CRDM Vent Shroud Support Assembly
- Reactor Vessel Head Insulation
- Equipment Qualified under 10CFR50.49
- Other Equipment in Containment

In summary, FPL has performed an extensive inspection to identify components which may have been affected by the boric acid from the conoseal leakage and has either replaced the affected components or determined that they are acceptable for use.

FPL, aided by technical consultants, including the original reactor vessel manufacturer, nuclear steam system supplier, and plant architect engineer, has evaluated the as-found condition of the plant to determine if the reactor coolant system pressure boundary had been reduced below its design basis or if the operability of equipment and components required for safe shutdown had been degraded beyond their design bases as a result of the leakage from the conoseal. As a result of that evaluation, FPL concluded that at no time was the unit in an unsafe condition as a result of the conoseal leakage. FPL has also analyzed the potential safety consequences if the leak had not been detected and had continued until the next Unit 4 refueling outage in March 1988.

This analysis has concluded that the limiting component failure due to corrosion is the conoseal carbon steel clamp (specifically, the closure bolt on the clamp). Such a failure would cause leakage in excess of Technical Specification limits which thereby would require operator corrective action before significant wastage could occur on the vessel head or adjacent components. Thus, operation of the unit until March 1988 would not have resulted in a condition beyond the design basis of the unit.

FPL also performed an investigation to identify the potential leakage mechanisms associated with the conoseal leakage. As a result of these inspections, FPL determined that the clamp shim and the conoseal gasket had significant damage and may have been associated with the conoseal leak mechanism. Based upon this information, it was determined that the two most likely potential leakage mechanisms were corrosion of the shim due to unidentified leakage from an external source and debris or imperfections in the conoseal; however, the existence of either or both of these potential leakage mechanisms could not be confirmed. In any case, once the lower conoseal leak initiated, it was probably exacerbated by corrosion wastage of the clamp and shim. FPL is taking actions to address these and other mechanisms, including 1) changing procedures and training of maintenance and inspection personnel and 2) modifying the thermocouple column assembly to provide for, among other things, the use of iron based superalloy clamps which do not use shims and which are not subject to any significant corrosion by boric acid. These steps will help prevent recurrence of leakage of the conoseal.

In addition to these actions, Turkey Point is taking steps to strengthen the detection and technical review processes associated with leaks. These steps will provide the added emphasis to leak detection, repair, and evaluations necessary to prevent recurrence of the type of problem which developed with the conoseal leakage.

Upon completion of the actions discussed in Sections 4 and 6 associated with preparation for startup, it is FPL's conclusion that the issues associated with this event will have been resolved and would not prevent return to normal operation.

2.0 INTRODUCTION

In March, 1987, during an inspection of the Turkey Point Unit 4 Reactor Vessel Head area, Florida Power and Light Company (FPL) found that a significant amount of boric acid crystals had been deposited on the Reactor Pressure Vessel (RPV) head due to a leak in the conoseal assembly on the northeast (NE) thermocouple column assembly. The purpose of this report is 1) to provide background information related to the conoseal leakage, 2) to identify the components which have been affected by the conoseal leakage and to describe the actions which FPL has taken for these components, and 3) to identify the potential

leakage mechanisms and to describe the actions which FPL has taken, or is planning to take, to prevent recurrence of this type of event.

The remainder of this report is divided into several sections. Section 3.0 describes the thermocouple column assembly, and the chronology of events leading to the discovery of the boric acid deposition on the reactor vessel head in March, 1987.

Section 4.0 describes the investigation which FPL has conducted to identify the extent of the components affected by the boric acid leakage, the impact of the boric acid on these components, and the actions which FPL has taken to assure that the affected components are repaired, replaced or can perform their intended function. Section 5.0 provides an analysis of the ability of the affected components to perform their intended safety function in their as-found condition in March 1987. Section 5 also provides an analysis of postulated continued operation with conoseal leakage.

Section 6.0 describes FPL's investigations to determine the potential leak mechanism associated with the conoseal leak and identifies the actions which FPL has taken or is planning to take to prevent recurrence of this type of event. Finally, Section 7.0 concludes that sufficient corrective action has been and is being taken in response to this event and states FPL's conclusion regarding return to normal operation.

3.0 BACKGROUND

3.1 Description of the Thermocouple Instrumentation Port Column Assembly

Turkey Point Unit 4 is equipped with incore thermocouples which measure fuel assembly coolant outlet temperatures at preselected core locations.

As shown in Figure 3-1, the thermocouples penetrate the reactor vessel closure head by utilizing four instrumentation port column assemblies. These instrumentation port column assemblies are pressure-retaining vessel appurtenances which can be disassembled to permit removal of the reactor vessel head from the reactor vessel for activities such as refueling.

As shown in Figure 3-2, each assembly consists of a lower and upper seal joint, each of which use a conoseal gasket for leak tightness. The lower seal is between a male flange and female flange, which is threaded and seal welded to the reactor vessel head penetration. The upper seal is between the male flange and a thermocouple support column. Each conoseal gasket is used only once and is replaced with a new gasket each time the thermocouple column is re-assembled.

Figure 3-1: General arrangement of instrumentation port column assemblies

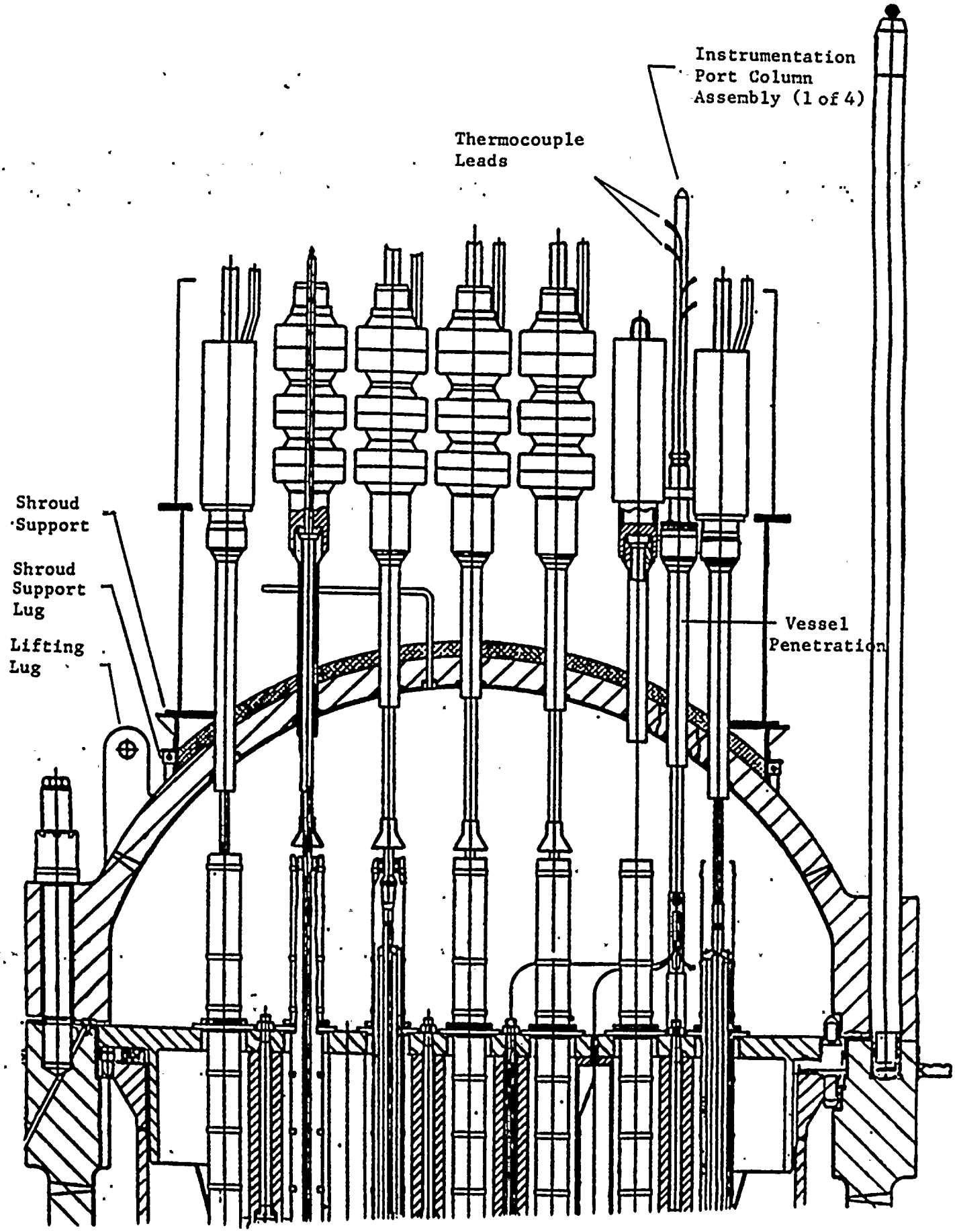
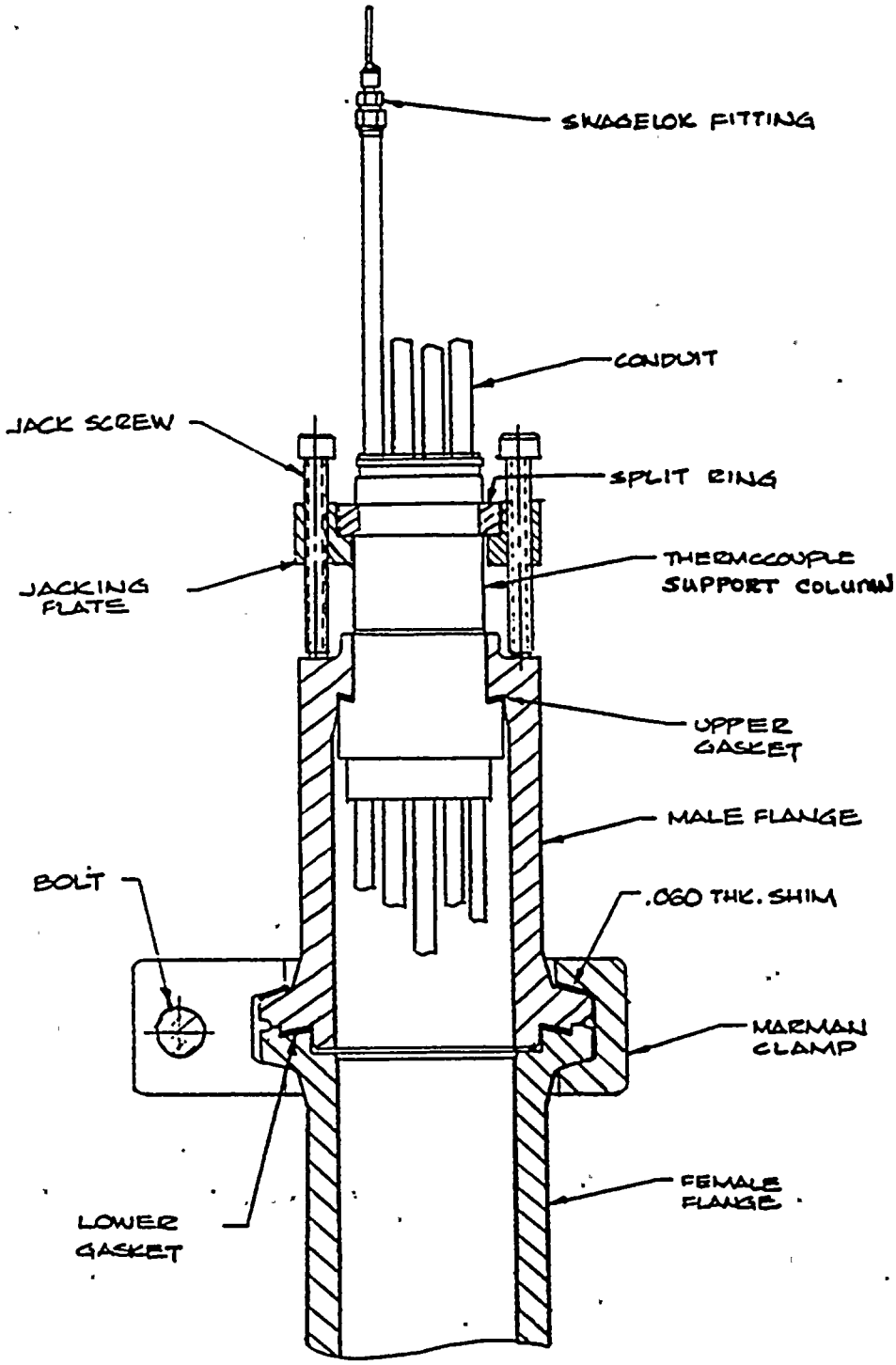


Figure 3-2: Instrumentation port column assembly



Assembly of the seals is performed after the reactor closure head is lowered onto the reactor vessel and the closure studs have been loaded through first pass loading. The steps in assembling the seals are as follows: First, the lower conoseal gasket is placed on the female flange, and the upper conoseal gasket is placed on the thermocouple support column. The male flange is then lowered over the thermocouple column and positioned on the female flange. A special fixture and hydraulic loading device is used to preload the lower gasket. The hydraulic pressure and resulting force is maintained on the assembly while a marman clamp is installed over the mating hub formed by the female and male flanges. The nuts on the clamp bolts are then tightened by hand. Three of the four lower thermocouple column clamps use shims (NE, SW, SE) to properly apply and hold the preload. Next the loading device is removed. The upper conoseal joint employs six jack screws to raise the thermocouple column and seat the upper conoseal gasket. These six jack screws are torqued in a prescribed sequence which is repeated at several intermediate torque values until the final prescribed torque value is reached. The nuts on the lower conoseal clamp are also torqued through several intermediate values to the final prescribed torque value. A lock wire is then installed in each pair of adjacent screws.

The conoseal gasket and the male and female flanges are stainless steel. The clamp assembly is made of carbon steel.

3.2 Chronology of Conoseal Leak

Conoseals have been used in Turkey Point Units 3 and 4 since they began operation. From the beginning, the four conoseals on Unit 4 (including the NE conoseal at issue here) used shims with the lower conoseal clamps to provide a proper fit on the male and female flanges. One conoseal clamp was replaced in 1982; no shim was required in the replacement clamp. During the 1984 refueling outage at Unit 4, a new shim was fabricated for the NE conoseal out of what Maintenance personnel have described as carbon steel. That shim was also used by the Westinghouse personnel who reassembled the conoseal in March 1986, the last reassembly before the conoseal leak was discovered in August 1986.

At the end of each refueling of the Turkey Point units, the primary coolant system is placed in an over-pressure condition (100 psi above normal pressure) and a visual walkdown inspection is performed on the primary system to identify any leaks. On August 12, 1986, such an inspection was performed on the Turkey Point Unit 4 reactor head and flange area, including each of the four conoseals. No leakage was observed by the inspector. Since the reactor coolant system was in an over-pressure condition and was at normal operating temperature, significant leakage out of the conoseal should have been visible as steam and/or spray.

From August 12, 1986 to August 30, 1986, the primary coolant system underwent two cycles of pressure and temperature change due to testing and corrective actions unrelated to the conoseals.

On August 30, 1986, preparations were made to bring the reactor critical. As part of the preparations for going critical, the FPL Quality Control (QC) Department performs a pre-critical containment walkdown. In preparation for this QC inspection, a Mechanical Maintenance foreman performed a pre-critical containment walkdown and discovered the steam leak from the NE conoseal. Upon notification, Management reinspected the leak and determined that an engineering evaluation was warranted prior to restart.

Power Plant Engineering was requested to observe the leak and to provide an evaluation as necessary to determine if the leak was acceptable for continued operation or if the leak required some corrective action before operation could begin.

After performing an inspection, Engineering prepared a safety evaluation which addressed the following factors:

- Amount of the leakage - The total reactor coolant system (RCS) leakage at the time was determined to be .45 gpm (with the conoseal leakage being only a minor contributor to the total amount). This amount of leakage was less than the limits in Technical Specification 3.1.3, which states that the reactor must be shutdown if the RCS leakage exceeds 10 gpm and that an investigation and evaluation must be conducted if the RCS leakage exceeds 1 gpm.

- Potential for Increases in Seal Leakage - Since the leaking joint was sealed with a stainless steel gasket, leakage was not expected to increase drastically in a short period of time, as would be expected in cases where a gasket of softer material is used.

- Potential for Corrosion of the Clamp - Based upon a corrosion rate of 30-50 mils/yr (the rate of corrosion of carbon steel in sea water, which was believed to be conservative) corrosion of the carbon steel clamp was estimated to amount to less than 1/32 inch during the next six months.

- Monitoring for Increases in Leakage - During operation, the RCS leakage rate is calculated once per day (or once per shift if leakage is greater than 0.5 gpm). If RCS leakage were to increase to greater than 1 gpm, an additional evaluation would be required pursuant to Technical Specification 3.1.3.

- Determination of Type of Leak - The leak was determined not to constitute a system fault such as a pressure boundary fault which would require an immediate shutdown by Technical Specification 3.1.3.

Based upon these other considerations discussed in the safety evaluation, Engineering concluded that the leakage was acceptable, and specified that another inspection of the clamp for corrosion shall be completed within six months (i.e., by February 28, 1987) and that the flange and clamp assembly should be disassembled and repaired during the next available shutdown. The engineer who performed the safety evaluation was not aware of the use of a shim in the leaking conoseal because the shim had not been indicated on any design drawings. The engineering evaluation was reviewed and its conclusions were accepted by members of the Plant Nuclear Safety Committee (PNSC). Accordingly, Unit 4 was taken critical on August 31, 1986.

FPL continued to monitor RCS leakage in accordance with procedures. Figure 3-3 identifies the RCS leakage of Unit 4 from August 1986 to March 1987, calculated in accordance with the methods provided in NUREG-1107. As this figure indicates, the calculated leak rates were scattered during this period, primarily due to uncertainties in the calculational methods and changes in plant operating conditions. Of importance, Figure 3-3 shows that there was no evidence of an increasing trend in RCS leakage during this period. Although the calculated RCS leakage did exceed 1.0 gpm on several occasions during this period, in each case FPL evaluated the leakage as required by the Technical Specifications and determined that the specific occasional increases in the calculated leakage was attributed to causes other than the conoseal leakage (e.g., identified packing leakage, etc.).

Similarly, other plant indicators did not experience any unexplained increase during this period which could be attributed to conoseal leakage. For example, Figure 3-4 shows that the flow rates on the sump did not increase from November 1986 to March 1987. Although the sump flow rate did increase in September and October 1986, this increase was attributable to the fact that the containment atmosphere in the earlier months was not yet saturated and at equilibrium temperature (i.e. leakage would have a greater tendency to remain in the containment atmosphere rather than to condense as a liquid and flow to the sump). Additionally, as shown in Figure 3-5, the applicable radiation monitor did not register any increase in radiation levels, with the exception of an increase attributable to higher RCS activity. Therefore, plant personnel had no indication of any increase in the conoseal leakage prior to March 13, 1987.

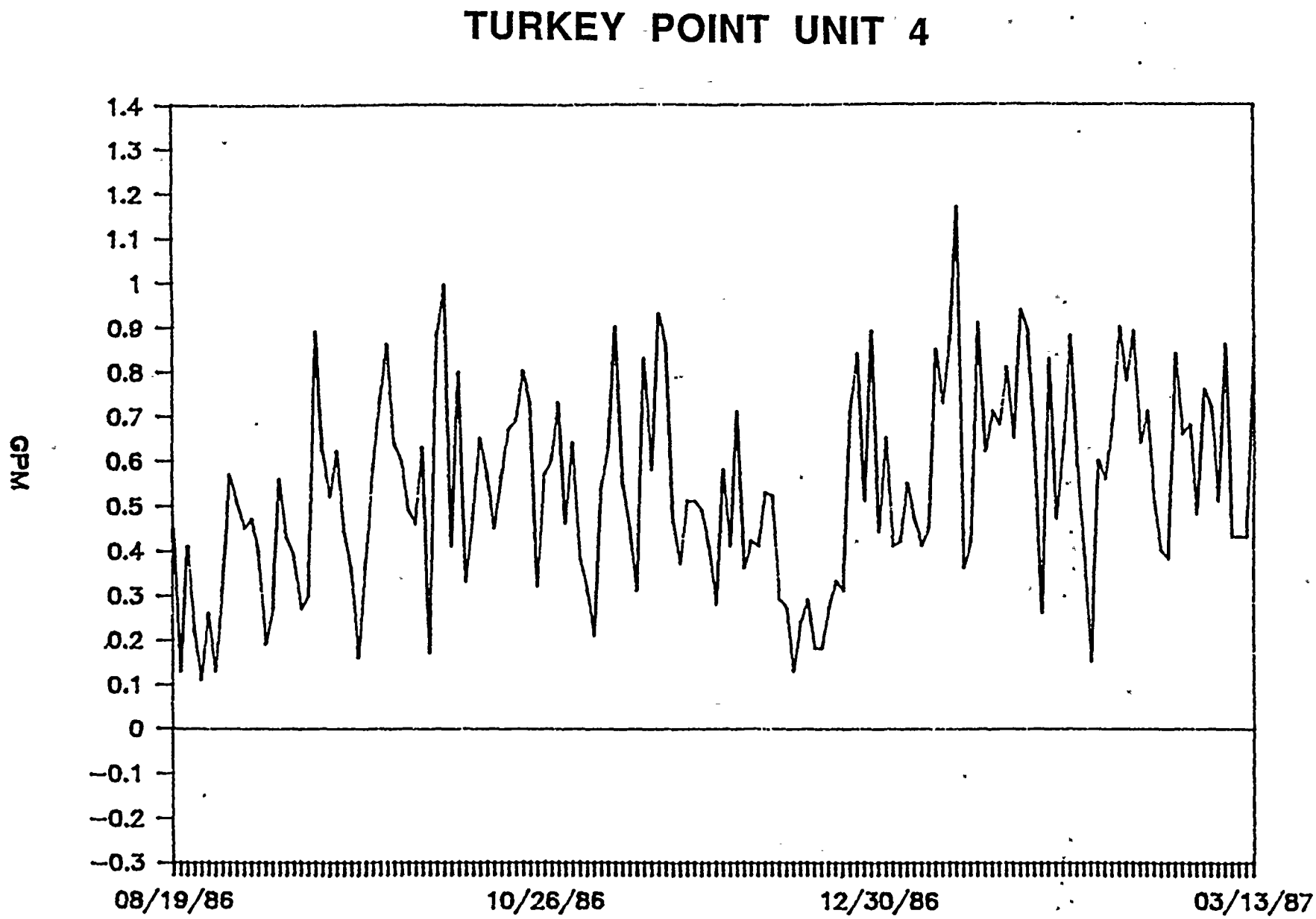


Figure 3-3: Unit 4 RCS leakage

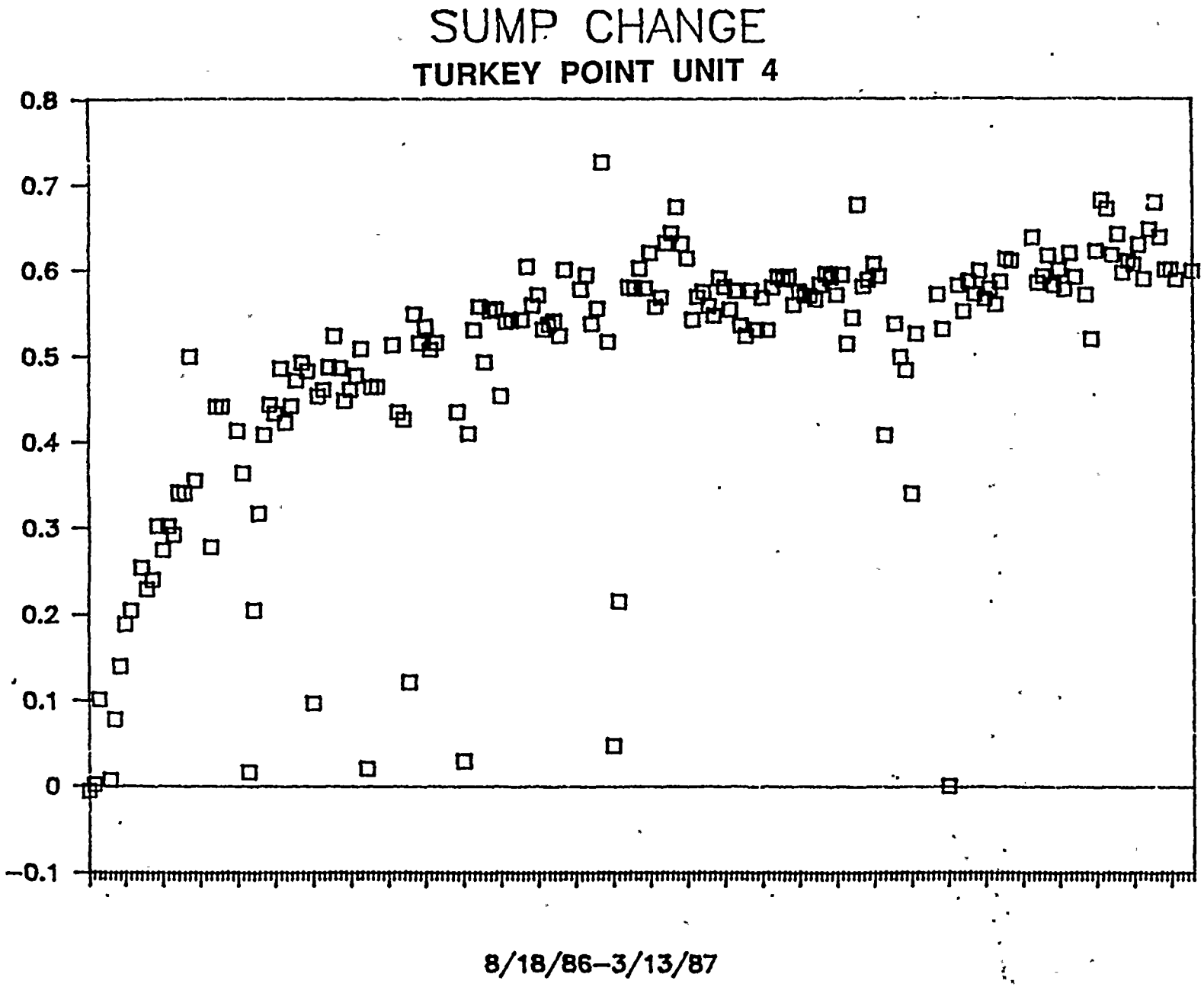


Figure 3-4: Unit 4 RCS Sump Change

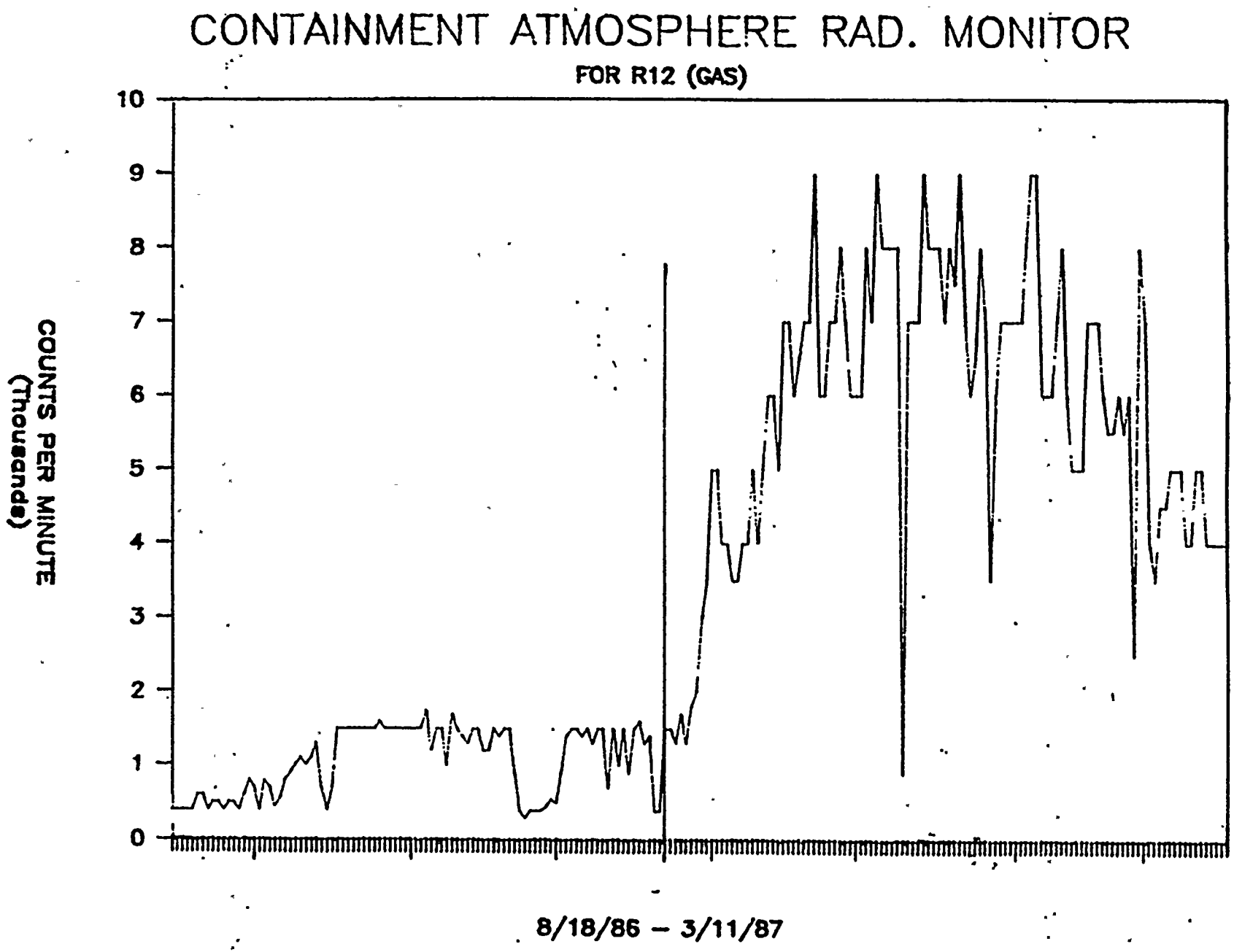


Figure 3-5: Unit 4 Containment Atmosphere Radiation Monitors

On October 24, 1986, Unit 4 was in hot shutdown for unscheduled condenser maintenance. The RCS pressure and temperature were 340 psig and 320 °F. Engineering took this opportunity to conduct another inspection of the conoseal leak. The leak rate did not appear to be greater than the rate observed during the inspection performed on August 30, 1986 (i.e., intermittent steam vapor puffs). Approximately one cubic foot of boric acid crystals was removed from the area of the instrumentation port column assembly and adjacent areas prior to Engineering performing its final inspection. No significant external corrosion or pitting indications were noted on the clamp or the visible bolting. Based upon this inspection, Engineering considered that the August 30, 1986 evaluation was still valid. Engineering also recommended that the clamp be disassembled and repaired during the next available shutdown of sufficient length.

On February 24, 1987, due to a request by the Plant Technical Department, Engineering commenced a review of the August 30, 1986 evaluation to determine if the date for inspection of the conoseal could be extended beyond February 28, 1987. In order to confirm the conclusions in the August 30, 1986 evaluation, FPL contacted Westinghouse on February 26, 1987 to obtain additional information on boric acid corrosion rates of carbon steel and experience with leakage of conoseals. Westinghouse did not immediately provide the information sought by FPL on February 26, 1987. Since information was sought from Westinghouse only to corroborate the corrosion rate used in the August, 1986 evaluation, Engineering did not believe it was necessary to await receipt of this information prior to completing its review. Therefore, based upon the acceptable results of the October 24, 1986 inspection and the August 30, 1986 evaluation, Engineering concluded on February 26, 1987, that the next inspection of the conoseal leak could be performed within six months of the October 24, 1986, inspection (i.e., by April 24, 1987). This conclusion was communicated to the Plant Technical Department via a memorandum.

Engineering continued to pursue confirmation of the conclusions. On March 13, 1987, Westinghouse responded to FPL's request for corrosion rate information. The information from Westinghouse indicated that the 30-50 mils/yr corrosion rate assumed in FPL's August 30, 1986, evaluation may not have been conservative. At the time this new information was received by FPL, the unit was in hot shutdown having been shutdown since March 10, 1987 for an unrelated reason. Upon learning this information, FPL brought the unit to cold shutdown, performed an inspection of the conoseal leak and found that a substantial amount of boric acid crystals was deposited in the areas of the thermocouple penetration and reactor vessel head. FPL notified the NRC of this condition, and the NRC sent an Augmented Inspection Team (AIT) to the site.

4.0 EVALUATION OF AFFECTED AREAS

After discovery of the boric acid deposition on the reactor head area, FPL initiated inspections to identify the items subjected to the deposition and to define the extent of damage. The first inspections were performed on items in the immediate vicinity of the leak, including items associated with the reactor vessel pressure boundary, the thermocouple column assembly, the conoseal, the Control Rod Drive Mechanisms (CRDM) and related equipment. Subsequent inspections included areas where boric acid could have been transported via the CRDM cooling system. FPL conducted walkdowns and analysis of equipment in the containment which is environmentally qualified under 10CFR50.49 to determine whether any of this equipment had been impacted by deposits of boric acid. Finally, FPL performed a more general walkdown of equipment in the containment to identify any other components or structures which may have been affected by the conoseal leakage.

Following these walkdowns and inspections, FPL took several actions for those items which had evidence of boric acid deposition. In general, these actions consisted of the following: The condition of the items was noted and the items were cleaned to remove any remaining boric acid deposits. Visual inspections and, as appropriate, Non-Destructive Examinations (NDE) were conducted for the cleaned equipment. The results of these inspections and examinations were then subject to an evaluation to determine whether to replace the affected items or whether the items could perform their intended safety function without replacement.

Conventional cleaning practices, such as steam cleaning, water washing, scraping, wire brushing or vacuuming were used with caution to preserve information for investigative reasons. Removal of the boric acid by cleaning of surfaces will prevent long term effects on materials and restrict any degradation of the materials to that which is present following cleaning. In the event small amounts of boric acid remain following the cleaning process, no significant effects on the carbon steel material will result.

Inspections were performed by teams which included representatives from the Engineering and Quality Control Departments. Quality Control Inspection Reports (IRs) were written to document the condition of the various components and the initial effect of the boric acid accumulation. When the results of these inspections indicated a need for corrective action beyond routine maintenance, a Non-Conformance Report (NCR) was issued.

The following sections discuss in detail each of the items which was inspected, the impact of the boric acid leakage on the items, and any corrective action for the items.

4.1 Reactor Vessel, Closure Head and Appurtenances

An initial cleaning and visual exam of the reactor vessel head was performed while the head was still on the vessel.

Several areas of the head were noted to be affected by the boric acid accumulation, including a segment of the head dome originating at the instrumentation port column penetration below the NE conoseal and extending down the head to studs 24 through 26 and down the side of the vessel flange and partially filling the head to vessel flange closure area. Boric acid accumulation partially filled the vessel flange to seal ledge annulus and spilled below onto the vessel insulation and the A loop hot and cold leg nozzle insulation. Boric acid also dripped onto the reactor vessel supports for the A hot and cold leg and into the primary shield wall penetration for the loop A hot leg nozzle. The boric acid also accumulated on the closure head flange knuckle region and ligaments from studs 5 through 37, around those studs, nuts and washers to varying depths and inside the vent shroud support ring in the vicinity of the leaking conoseal.

Detectable wastage was observed in the instrumentation port column penetration counterbore (No. 53), a small area on the head base metal adjacent to the penetration, a three foot arc of vent shroud support, a vent shroud support lug, studs 24, 25 and 26, nuts 23 through 28, washers and small areas on the head flange ligament between the washers. (See Figure 4-1)

Following the removal of the closure head to the head storage stand, a visual examination was made, the affected areas were further cleaned, and additional detailed inspections were made on this and various other components as discussed in the following sub-sections. The detailed inspections utilized standard methods of NDE including Ultrasonic Testing (UT), Magnetic Particle Testing (MT), and Visual Testing (VT).

4.1.1 Closure Head Dome and Flange

These components were inspected to determine the depth of penetration and extent of the area affected by corrosion as a result of exposure to the boric acid accumulation. An initial visual examination of the reactor vessel closure head dome and flange while the head was still on the vessel revealed localized surface wastage and loose rust scale on the head flange and dome as discussed below.

Two areas, each less than 4 square inches and located in the ligaments between studs 23 and 24 and between studs 24 and 25 on the top of the head flange, had wastage to a maximum depth of 1/16 inch. An evaluation of the two depressions in the ligament top surface region concluded that the condition is acceptable because the critical

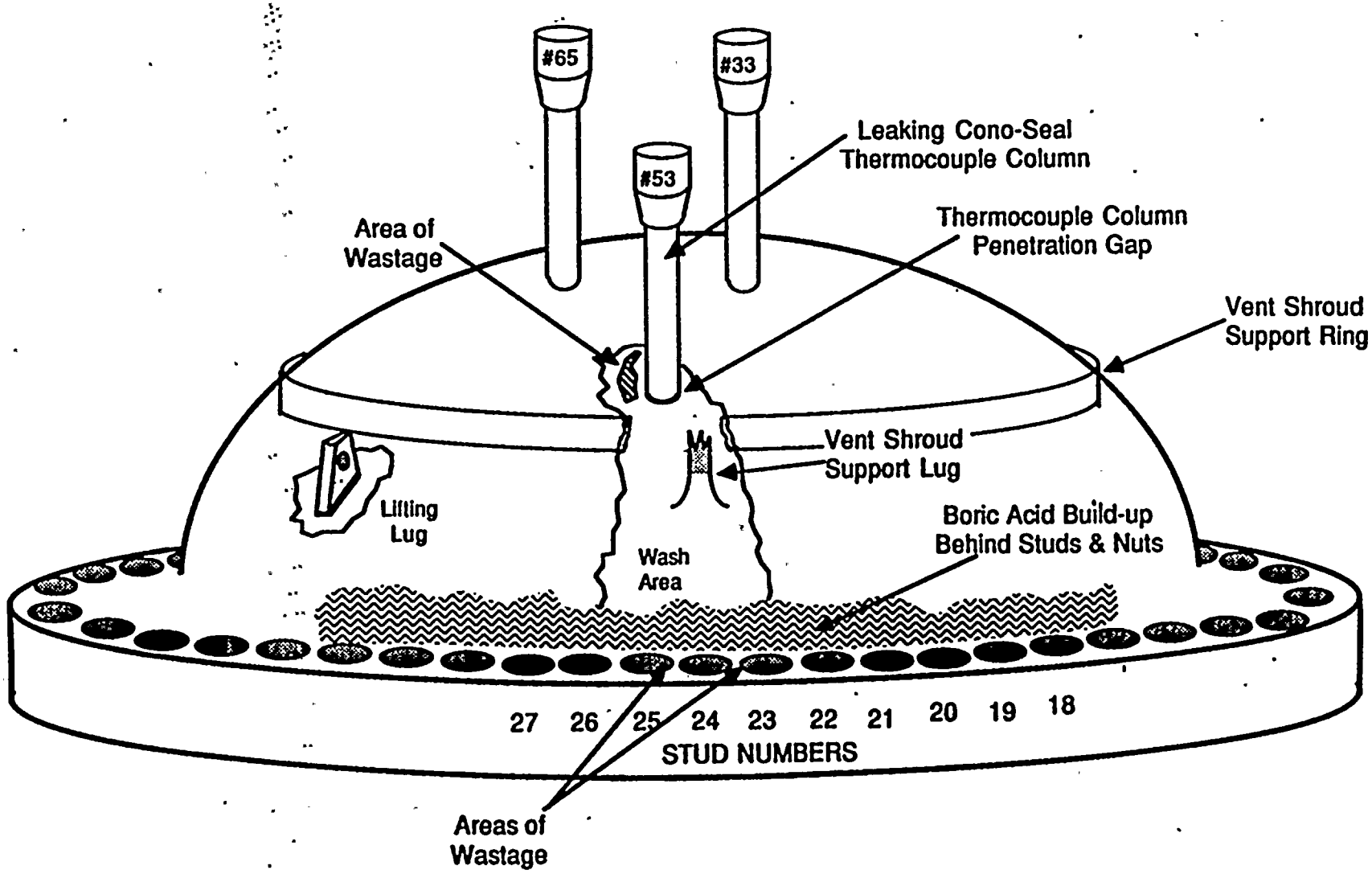


Figure 4-1: Identification of areas on the reactor vessel head affected by boric acid

flange region is beneath the stud nuts and washers and not at the top of the ligament region (which is not highly stressed). Since the boric acid accumulations have been removed, the remaining stains will have no adverse effect on continued operation.

The closure head flange vertical outside diameter and bottom surface including the mating surface and stud hole region exhibited no significant corrosion or indications, showing only slight staining in the area of the boric acid buildup. Visual and video inspections of the stud holes also showed staining at the 90° quadrant closest to the vessel centerline on some stud holes. However, there was no corrosion or erosion present.

The closure head dome was inspected by UT for thickness in the areas affected by the boric acid accumulation. The results of these exams indicated that the head thickness varied from 6.8 to 7.0 inches. These thicknesses are still well in excess of the minimum head thickness of 6.1875 inches specified on the reactor vessel design drawings. The head to flange weld between stud holes 5 and 37 was UT and MT examined and revealed no indications.

The closure head and flange areas affected by the boric acid were given a surface exam by MT. This exam did not reveal any unacceptable indications on the flange, dome or lifting lug under the applicable provisions of Section XI of the 1980 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code through Winter 1981 Addendum. The closure head was given a detailed visual examination in the area of the leaking conoseal where the permanent dome insulation was damaged and displaced. This exam revealed a boomerang shaped depression eroded into the head dome with one leg approximately 8-1/2" long, and the other leg approximately 4" long and 1-1/4" wide at its widest point. The depression was located in the ligaments between the leaking conoseal (Penetration No. 53) and Penetrations No. 65 and No. 33. The edge of the depression was situated approximately 3/8" to 1/2" from Penetration No. 53 and was 1/4" deep at its deepest point. This depression is shown in detail on Figure 4-2. As stated earlier, the subsequent MT examination of this depression showed no other surface defects.

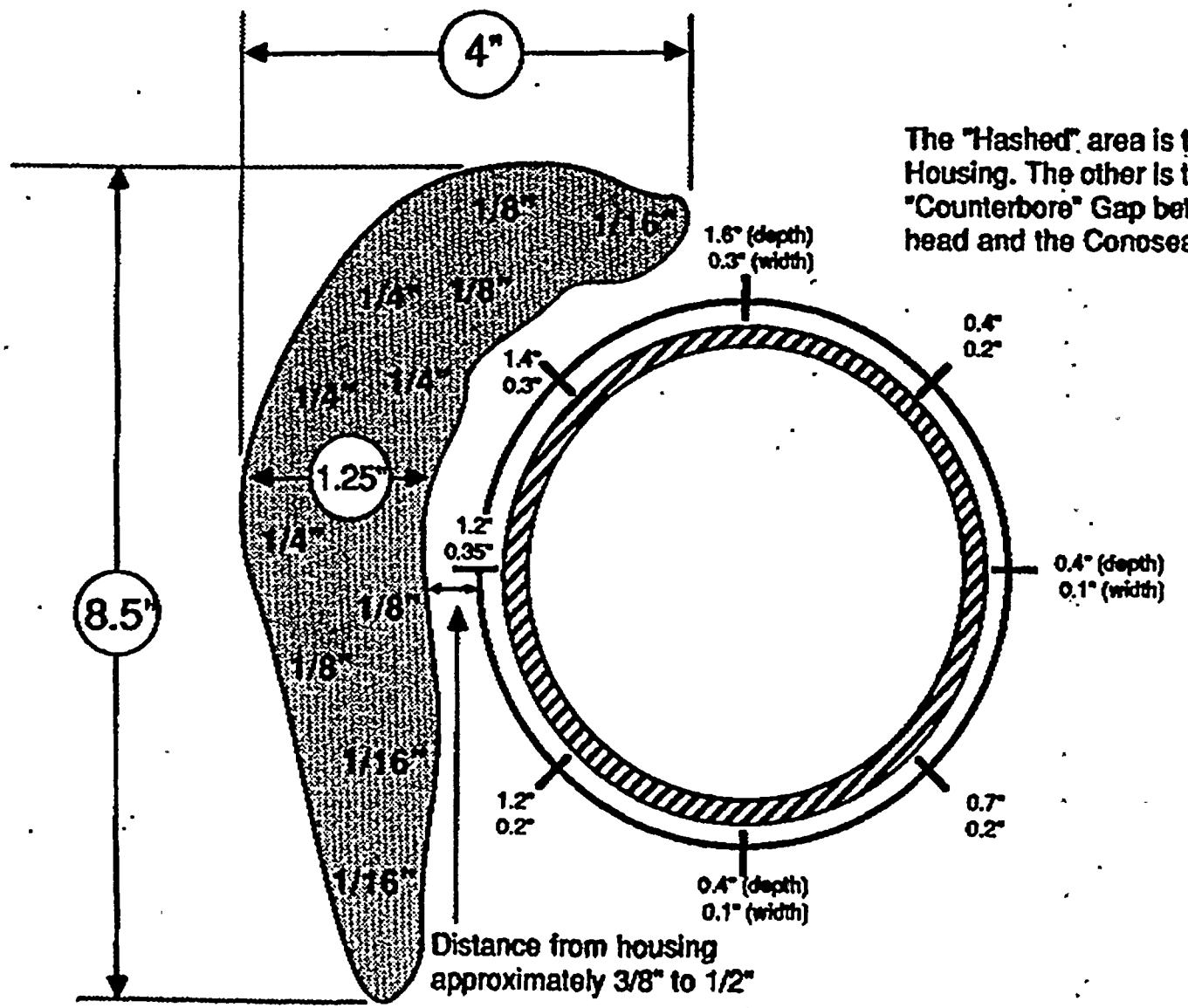
An evaluation, in accordance with the ASME Code, was made of this depression and its effect on the minimum wall thickness of the head and of the reinforcement of the enlarged hole in the head at Penetration No. 53. This enlarged penetration hole is further discussed at the end of this section. The results of this evaluation showed that the ASME Code sizing and reinforcement requirements are met with this depression present. Since the closure head dome insulation was damaged in the vicinity of Penetration No. 53, FPL surmises that the insulation sealer had lost integrity and the Kaowool insulation had become saturated with condensate and boric acid. The hot closure head dome and head adapters could then have been in contact with the condensate during plant operation between August 30, 1986 and March 13, 1987. Boiling of the condensate in intimate contact with the closure head would have transferred heat away from the location, causing a local cold spot on the head dome inside the ventilation shroud. The location could have been constantly wetted. This local cold spot would cause a radial thermal gradient across the head thickness and alter the head stresses at that location. An evaluation was performed in accordance with Paragraph NB-3228.3 of the ASME Code, Section III to determine the effect of this postulated local cold spot. The analysis concluded that the reactor vessel head dome with the postulated cold spot is acceptable under the ASME Code.

There was boric acid build-up to varying depths on the head dome insulation as the flow moved away from Penetration No. 53. For conservatism, FPL has removed and replaced all of the dome insulation and inspected the total head dome. This inspection revealed no further damage to the head.

The instrumentation port column Penetration No. 53 associated with the leaking conoseal was inspected. The counterbore of the penetration had been enlarged by wastage from 0.005 inches to a maximum of 0.35 inches at the top of the head. The maximum depth of the counterbore gap corrosion was measured to be 1.6 inches. (See Figure 4-2).

Figure 4-2: Detail of vessel head dome wastage and Penetration No. 53 counterbore corrosion

The sketch below is a plan view as looking down on the penetration. The numbers circled are dimensions, the others are depth measurements.



The "Hashed" area is the Conoseal Housing. The other is the "Counterbore" Gap between the head and the Conoseal housing.

A calculation was performed which verified the acceptability of this condition under the ASME Code Section III criteria for the head minimum thickness, ligament efficiency, and penetration reinforcement requirements.

All the remaining head penetrations and their associated counterbores in the head were visually examined with no evidence of corrosion noted. Based on the observations and evaluations performed in this section, it has been determined that the reactor vessel closure head is suitable for continued safe operation.

4.1.2 Reactor Pressure Vessel Head Studs, Nuts and Washers

Reactor pressure vessel closure head studs, nuts and washers, No. 5 through 37, were visually examined after removal of boric acid residue from the flange and closure area. These studs were in the area of boric acid deposition and were determined to require further investigation, evaluation and disposition. Wastage rather than stress corrosion cracking is the principal concern with respect to the studs. High yield strength (150-175 ksi) quenched and tempered martensitic steels such as the 4340 stud material have shown resistance to stress corrosion cracking in boric acid solutions even when samples were pre-cracked and highly loaded. Only three studs directly below the leak (Nos. 24, 25, 26) were observed to have noticeable wastage of the threaded area above the nuts. In addition, nuts No. 23 through 28 and large and small washers No. 24 through 27 were observed to have wastage.

In an effort to determine if wastage had occurred between the stud and nut interface, between the head flange to washer interface, between the washer to washer interface or between the washer to nut interface, it was decided to utilize the manual ultrasonic examination technique looking from the inside surface of the stud. This examination was conducted on stud Nos. 22, 23, 24, 25, 26, 27, and 28 within the region of boric acid deposition and stud Nos. 2, 46 and 54 outside the region of boric acid deposition. The results of these examinations identified anomalies at the bottom of the nut between the nut and the washer on studs 22, 23, 24, 25, 27, and 28 (i.e. those within the region of boric acid deposition). No anomalies were noted on studs 2, 46 or 54 (i.e. those outside the region of boric acid deposition.)

These results showed that only the studs within the area of boric acid exhibited anomalies when ultrasonically

examined. A decision was made to remove studs No. 5 through 37 from the head and to conduct a full 100% ultrasonic and magnetic particle examination in accordance with the ASME Code.

The results of these examinations indicated that studs No. 5 through 23 and studs No. 27 through 37, nuts No. 5 through 22 and nuts No. 29 through 37, large washers No. 5 through 23 and No. 28 through 37, and small washers No. 5 through 23 and No. 28 through 37 were acceptable for continued service. FPL conservatively decided to replace studs, nuts and washers No. 22 through 28 inclusive.

4.1.3 Reactor Vessel Flange and Stud Holes

The top of the reactor vessel flange was visually inspected during the head removal operation. The bolting ring was very rusty in appearance but there was no apparent material loss.

The stainless steel mating surface was free of boric acid indications except for loose flakes and particles at the outer perimeter. This debris had most likely fallen onto the mating surface from the bolting ring and closure head during the lift. The flange was then cleaned, visually inspected and determined to be satisfactory.

Stud holes No. 24, 25, and 26 were inspected. This inspection consisted of a review of a video tape which was prepared by means of lowering a video camera through the closure head stud hole into the corresponding vessel flange hole. Boric acid accumulation was observed within the top three inches of stud hole No. 25. Boric acid was also observed on the upper one inch of thread in stud hole No. 24, and the boric acid was only sitting on the top thread at the base of the counterbore in stud hole No. 26. After cleanup of the deposits and reinspection, FPL concluded that the threads were in good condition and acceptable for continued use. No thread or counterbore damage was found in any other flange hole during visual inspection performed during stud hole cleaning.

4.1.4 Annulus Region and Reactor Vessel Shell, Insulation, Nozzles and Nozzle Supports

The annulus between the reactor vessel and cavity liner from the seal ledge elevation to the Nuclear Instrumentation (NI) thimble support ledge was inspected. This inspection covered the reactor vessel shell insulation, hot and cold leg nozzle insulation, vessel supports, NI detectors and wells, thimble support ledge and the cavity liner.

The upper annulus is defined as the elevations bounded by the seal ledge and the top of the nozzle penetrations through the primary shield wall. Inspection of this region revealed no boric acid buildup or damage of the seal ledge vertical or underside horizontal surfaces between studs No. 20 and 30. Some staining and boric acid buildup was noted on the reactor vessel insulation immediately (1") below the interface of the reactor vessel insulation and the underside horizontal surface of the seal ledge between studs No. 20 and 30. There was no indication that any leakage had penetrated the interface, thus the insulating capability of the reactor vessel insulation was maintained. The condition of the upper annulus regions bounded by studs No. 5 through 13 and No. 30 through 37 was normal. The upper annulus regions bounded by Studs No. 14 through 21 and No. 25 through 29 exhibited staining on the reactor vessel insulation and boric acid buildup of 1/2" to 6" on the cavity horizontally approximately 26 inches below the seal ledge. The region bounded by studs No. 22 through 24 contained heavy boric acid buildup on this ledge reaching to within 4 inches of the seal ledge, completely filling the air gap between the reactor vessel insulation and the cavity wall.

Minor boric acid buildup was found on the top surfaces of the "A" hot and cold leg nozzle insulation with some staining evident on the sides. Reactor vessel supports at the "A" hot and cold leg nozzles were found to have boric acid buildup of approximately one inch on the horizontal surfaces. The condition of "B" hot and cold leg nozzle insulation and the associated supports was normal. Boric acid buildup was identified in the "A" cold leg shield wall penetration.

Inspection of the lower annulus, elevations below the reactor vessel supports bounded by studs No. 16 and 31, revealed minor boric acid staining on the reactor vessel insulation. The NI detectors, located above the thimble support ledge, exhibited boric acid buildup on the vertical surfaces. Slight boric acid residue was found on the detector well vertical surfaces. Boric acid deposits up to one inch thick were found on the thimble support ledge.

With the exception of the NI detectors (which were replaced), engineering review of these inspection results recommended thorough cleaning of all affected areas followed by further inspections. Prior to restart, subsequent inspections will be used to verify that these components will have no significant corrosion from boric acid deposition and will be acceptable for use.

Unacceptable components will be replaced or repaired as appropriate.

4.2 Instrumentation Port Column Assembly

The conoseal (NE location) was inspected in order to identify the condition of the assembly, and in particular to identify any abnormal condition in any parts of the assembly.

The assembly had heavy deposits of boric acid crystals on the external surfaces. After removal of the crystals, there was evidence of corrosion on the external surfaces of the clamp. The mechanical clamp was determined to be intact and tight. The lockwire was cut and the jack screws were tight with some boric acid present. During disassembly, some deterioration was noted on the clamp nuts. All three clamp bolts exhibited surface corrosion. Additionally, one bolt was measured to have a localized reduction in neck diameter of approximately 11%. The inside surface of the clamp ring showed signs of erosion in the central area around the entire circumference. The depth of erosion varied from approximately .061" to .218". The spacer (shim) utilized in this location was severely deteriorated and found in two pieces.

Following removal of the male flange, inspections revealed that both the upper and lower conoseal gaskets were intact and properly oriented. Only discoloration of the upper conoseal gasket was noted, whereas the lower conoseal gasket had scratches, nicks and crimps that were later determined to be due to the tool used to remove the male flange. At one location on the lower conoseal gasket, a blue discoloration about half the size of a dime was present. Cleaning of the conoseal gasket revealed that the blue spot extended to the outer seating surface and erosion to that surface was evident. Upon inspection of the female flange, a similar blue discoloration was observed that is assumed to correspond with the discoloration on the gasket. The male flange also exhibited a localized slight blue discoloration. Jack screws and other hardware associated with the upper conoseal showed no signs of any abnormal condition.

On March 24, 1987 the three (3) other conoseals were disassembled and no evidence of leakage or boric acid crystal accumulation was noted. Additionally, each of the upper and lower conoseals were found in good condition. Finally, no corrosive damage was noted on any other components of the column assemblies during the inspection.

Visual inspections of the NE conoseal female flange showed evidence of erosion as a result of the leak. FPL replaced the female flange with an identical replacement which has been positioned and welded in place.

In order to simplify the instrumentation port column assembly and also to minimize the assembly time required, an enhanced instrumentation port column assembly will be utilized at all four locations at Unit 4 as well as Unit 3. This assembly consists of a new male flange, a modified lower conoseal clamp and a modified upper conoseal clamp. The modified clamps are referred to as articulated and are designed to minimize the number of loose pieces.

The modified lower clamp consists of one integral part with no loose pieces. After proper positioning over the male and female flanges, the clamp is closed by rotating a single pivoted bolt into position and torquing the nut. As the nut is tightened, the lower gasket is seated, thus eliminating the need for a hydraulic preload device. At the upper conoseal, the jack screw configuration is being eliminated in favor of an upper articulated clamp. In order to provide a compatible interface with the upper articulated clamp, a new male flange design and positioner is necessary. The male flange design is only different at the top where the profile is modified to interface properly with the upper articulated clamp. The upper clamp provides a benefit in that the number of separate pieces is reduced and the need for lock wiring is eliminated. Like the lower clamp, the upper clamp is one integral part and secured by rotating a pivoted bolt into place and torquing the nut. Once the first nut is tightened, which seats the conoseal, a jam nut is installed.

The benefits associated with this modification package include reduction in radiation exposure time, reduction in critical path outage time, and a simplified assembly effort. No change in the conoseal material, sealing surface geometry or female flange design is associated with utilization of this modification. During plant start-up of Unit 4, the performance of all four instrumentation port column seals will be monitored and documented.

4.3 Control Rod Drive Mechanisms (CRDMs) and Rod Position Indication (RPI), Electrical Connectors, Cables and Instrument/Control Equipment in Locality of Leak

The possible effects upon the RPIs, CRDM coils, electrical connections, and instrumentation in the vicinity of the leak were investigated. This inspection covered CRDM cables and connectors, RPI cables and connectors, RPI coil stacks, CRDM coil stacks, CRDM cooling fan motor, Core Exit Thermocouples (CET), Heated Junction Thermocouples (HJTC) and cable trays in the vicinity of the head. Visual inspections as well as resistance checks were performed.

The visual inspection of the RPI coil stacks showed some boric acid residue on most of the stacks. No evidence of surface corrosion or material degradation due to the boric acid was noted. A check of the RPI coil stacks resistance values and insulation

resistance was performed, and the results were satisfactory. Also, no damage attributable to boric acid was found on the RPI electrical connections. The boric acid residue has been removed. Based on the above investigation, FPL determined that the RPI system has not been adversely affected by boric acid and the system was acceptable for use. A previously planned replacement of the RPI cables and electrical connections will be accomplished prior to start up.

A visual inspection of the CRDM coils revealed no significant amount of boric acid accumulation and no visible corrosive damage to the coils. The CRDM electrical connections were found not to be degraded by the presence of any boric acid. Tests were also performed on the CRDM coils measuring resistance values and insulation resistance. The results showed six coils to be outside the acceptable range of coil resistance; these will be repaired or replaced as appropriate. A previously planned replacement of CRDM cables and electrical connections will be accomplished prior to startup.

The HJTC system and the CET system have been qualified for post-accident monitoring per Regulatory Guide 1.97. The visual inspection of the HJTC connectors was performed with no physical damage noted. The protective caps were all properly installed. Likewise, the CET system was visually inspected and verified to be in good physical condition. Therefore, the HJTC system and CET system are acceptable for continued service.

CRDM and RPI cable trays that had been exposed to boric acid have been inspected and verified that no boric acid was present and no damage had occurred. Therefore, these cable trays are acceptable for continued service.

4.4 Control Rod Drive Mechanism Cooler Unit

The CRDM cooler units (4V2A and 4V2B) provide a means by which heat can be removed from the CRDMs, thereby providing a less severe operating environment with respect to temperature. The CRDM cooler units provide a non-safety related function; however, the heat exchangers within these cooler units are considered safety related since they are part of the Component Cooling Water (CCW) system.

Significant boric acid deposits were identified by visual inspection of the CRDM coolers. The CRDM coolers were disassembled, inspected and cleaned. Each component was inspected for corrosion due to contact with boric acid. No significant material deterioration or pitting was revealed. The structural integrity of all components has been maintained. Following testing, the CRDM cooler units were returned to service.

Inspection of CRDM A cooling fan motor showed no evidence that it was adversely affected by the boric acid spray. While CRDM B cooling fan motor was still operating when the Unit was shutdown, insulation resistance testing after cleaning identified a ground. The motors were electrically tested by measuring phase to ground and phase to phase resistance values. CRDM B cooling fan motor failed these tests and will be replaced prior to restart. The CRDM cooling fan motors will be acceptable for continued service after replacement of CRDM B cooling fan motor.

4.5 Control Rod Drive Mechanism Vent Shroud Support Assembly

The vent shroud support assembly is composed of a 1/2" thick cylindrical ring and flange bolted to three lugs which are welded to the surface of the reactor vessel head. The support ring is designed to support the shroud assembly independent of the ductwork leading to the CRDM coolers. The ring is fabricated from carbon steel, as are the lugs welded to the reactor vessel head, while the shroud is fabricated from stainless steel. The 3/4 inch diameter reactor vessel head vent pipe penetrates through, and is supported by, the shroud assembly.

Inspection results revealed that a three foot section of the shroud support in the vicinity of the leaking conoseal had been consumed by the boric acid. The welded support lug closest to the leak had been attacked by the boric acid and was no longer providing an anchor point for the shroud support due to wastage of the lug material.

Degradation of the CRDM vent shroud support assembly could result in movement of the shroud from its anchor position. This could affect the non-safety related CRDM cooling system and support of the safety related head vent pipe. To address the concern regarding over stressing of the head vent pipe, clearances were measured at the annular space between the shroud and the CRDM housings. Review of these measurements indicate a consistent spacing between the shroud and CRDM housings. This indicated that, although the shroud support was damaged, the damage did not result in shifting or movement of the shroud. Therefore, FPL concluded that no abnormal stresses were transmitted to the head vent pipe as a result of degradation of the CRDM vent shroud support. The remaining two support lugs were visually and magnetic particle inspected with no significant indications found.

The shroud support has been repaired by cutting out the damaged section and welding in a new section. The damaged portion was cut and removed at locations not affected by the boric acid. The new section was fabricated in accordance with the original support design configuration. The replacement material used to repair the damaged section is equal to or better than the original material and is compatible with the existing material. The damaged area of

the support lug was also repaired by removal and replacement in accordance with Section XI of the ASME Code.

In summary, repairs to the lug and support assembly have returned them to their original design and configuration. Following these repairs, the shroud to CRDM clearances were checked and proper alignments were verified to have been maintained.

4.6 Reactor Vessel Head Insulation

Inspection of the reactor vessel head reflective insulation revealed that four of the nine panels were contaminated with boric acid deposition in the form of crystals. In order to ensure that no future concerns arise from this contaminated insulation, all nine existing panels will be replaced with new panels which provide at least equivalent performance.

Inspection results of the reactor vessel head permanent insulation indicated that some sections of the permanent insulation came into contact with the leaking reactor coolant. In order to ensure that no concerns arise from this contaminated insulation, all the reactor head permanent insulation has been replaced with new insulation which provides at least equivalent performance.

Additionally, the replacement insulation meets the guidelines of Regulatory Guide 1.36.

4.7 10CFR50.49 Equipment Inspection Results

The 10CFR50.49 electrical equipment subject to exposure to the conoseal leak (i.e., inside containment) was analyzed for potential degradation as a result of boric acid exposure. As discussed below, the analysis consisted of a walkdown of 10CFR50.49 equipment inside the containment and a material analysis.

Approximately 20% of the equipment on the 10CFR50.49 (EQ) list located inside containment was inspected initially via walkdown. Based on observations made during this walkdown a procedure was developed to walkdown the remainder of 10CFR50.49 equipment inside containment.

The walkdown was performed after an initial cleaning of the containment. Visual inspections were made and verified, photographs were taken, and reports were written to document the walkdown team's findings. These reports were then forwarded to Engineering for review and analysis. The results showed that 149 of the 165 pieces of equipment exhibited no signs of boric acid deposition or corrosion. With the exception of cables, all 10CFR50.49 equipment that was identified as potentially affected by boric acid deposition will be cleaned and visually reinspected

to ensure that no deterioration as a result of boric acid deposition occurred.

With regard to electrical cables, FPL intends to develop a cleaning process to remove boric acid deposits or to confirm that degradation due to deposited boric acid is not a factor in the forty year life of these cables. In this regard, a conservative analysis has been performed which shows that no degradation will occur prior to August of 1988. During that time, cleaning will be performed or the forty year life of the cables without cleaning will be confirmed. In addition, the impact on flamastic from the boric acid was evaluated. It was concluded that no adverse impact on the flamastic would occur from contact with boric acid.

4.8 Other Equipment

In addition to the above special inspection and evaluation performed on the 10CFR50.49 equipment, a broader inspection was performed on those other areas in the containment which could possibly have been affected by the conoseal leakage. Inspection guidelines and documentation requirements were established by Engineering and QC. The areas inspected were chosen based on the flow path of the leaking conoseal and the proximity of equipment to the CRDM and Normal Containment Coolers (NCC) systems. These are the areas where boric acid deposition was anticipated.

The objective of these inspections was to determine the extent of boric acid deposition and to record the as-found condition. The general areas and components identified were:

1. Steam generator hot and cold leg penetrations,
2. Manipulator and polar cranes,
3. General area at top of pressurizer cubicle,
4. Reactor sump general area,
5. Ex-core detector wells and housings,
6. Elevation 14' general area outside the biological wall around "A" steam generator,
7. Elevation 14' general area inside the biological wall around "B" and "C" steam generators,
8. Elevation 14', other general areas,
9. Elevation 58', general area,
10. Elevation 30'6" general area at pressurizer cubicle,
11. Elevation 30'6", other general areas,
12. CRDM vent ductwork "donut" and risers, and
13. Reactor vessel missile shield.

The effects of the conoseal leak varied in degree and were confined to the expected locations. Boric acid residue from the conoseal leak was found in the following locations:

1. Reactor vessel loop A hot leg penetrations showed boric acid build up. Five remaining reactor coolant piping penetrations were free of boric acid deposition,
2. Reactor sump general area - Boric acid build up (1" thick) was found on the floor directly beneath studs 24, 25, 26, and on the cavity wall and reactor vessel insulation in the northeast quadrant,
3. Excore detector wells and housings - Minor residue was found in the general vicinity outside the wells for detectors N31, N35, N41 and the spare at 90°. No damage to the well cover seals was observed. A slight dusting of boric acid was found inside the housings and a large amount of boric acid was found on source range detector N31 and intermediate range detector N35, power range detector N41 and associated cables. Source range detectors N31 and intermediate range detector N35 have been replaced. N41 detector will be replaced prior to startup. The affected cables will be cleaned.
4. Elevation 14', general areas outside the biological wall - A buildup of boric acid was found on the floor and baseplates around the elevator and piping insulation covers overhead around the elevator. Deposits were also found on cable trays, conduit, supports, terminal boxes, small bore piping and on several valves,
5. Elevation 14', general area inside the biological wall - A build up of boric acid was found on the floor and in several baseplates near the "C" steam generator. Deposits were also found on the Reactor Coolant Drain Tank pump baseplate and anchor bolts,
6. Elevation 14', other general areas - Boric acid deposits were found on various valves, several baseplates and anchor bolts, small bore piping and tubing supports,
7. Elevation 58', general area - Boric acid deposits of varying thickness were found on the heat exchanger coils in the plenum and fan assembly on the Normal Containment Coolers (NCC) 4V1A,B,C,D. In general, deposits were found on the lower section of the heat exchanger coils and the lower support channel. The lower support channel was blocked by rust and boric acid deposits, which did not permit flow to the drip pan. This resulted in boric acid deposits overflowing from the drip pan to areas directly below the NCCs,
8. Elevation 30'6", general area at pressurizer cubicle - A light amount of boric acid buildup was found on the floor and base plates, conduit, cable trays and other miscellaneous equipment,
9. Elevation 30'6", other general areas - Boric acid residue was

discovered on ductwork, conduit and an electrical junction box,

10. CRDM vent ductwork "donut" and risers - The north section of ductwork show a path of boric acid residue back towards the riser. The south riser showed slight evidence of boric acid deposition, and
11. Reactor vessel missile shield - Boric acid deposits were found on the end of one of the I-beam supports for one of the two missile shield slabs.

5.0 SAFETY ANALYSIS

5.1 Safety Analysis of As-Found Condition

In order to determine if the leaking conoseal had, at any time, compromised the design basis of the plant, FPL has evaluated the as-found condition of the plant. The evaluation assessed the condition of the plant as-found to determine if the reactor coolant system pressure boundary had been compromised beyond the design basis of the plant or if the operability of equipment and components required for safe shutdown had been degraded beyond the design basis for the plant as a result of the leakage from the conoseal.

The as-found condition of the unit revealed the following six major areas that have been evaluated for a potential impact on plant operation. These areas were: 1) the loss of reactor coolant through the conoseal assembly, 2) the resultant corroded boomerang-shaped depression in the vessel head dome, 3) the enlarged penetration to vessel gap caused by corrosive attack, 4) the localized wastage on the vessel flange between studs 23-24 and 24-25, 5) the corrosion exhibited on several closure studs, nuts and washers and 6) the effect on the fatigue life of the vessel head due to the possibility of a local cold spot. All safe shutdown components were bounded by the safety evaluation. Each of the six areas of concern have been evaluated as discussed below.

5.1.1 Loss of Reactor Coolant Through Leaking Conoseal

At the time of plant shutdown, the total RCS leakage rate was calculated to be 0.45 gpm, with the conoseal leakage being only a portion of this total amount. Technical Specification leakage limits were not exceeded and the RCS volume was maintained by the normal plant make-up system. This indicates that the plant was well within the design basis for normal and transient operating conditions.

5.1.2 Wastage of Vessel Head Dome

The impact of this area of wastage was evaluated by performing a calculation in accordance with the ASME Code. Results of the calculation indicated that the design acceptance criteria for minimum wall thickness of the head were satisfied.

5.1.3 Enlarged Vessel Penetration Gap

Enlargement of the Penetration 53 gap due to corrosion was also evaluated by performing a calculation in accordance with the ASME Code. The design criteria for head minimum thickness, ligament efficiency and penetration reinforcement requirements were satisfied.

5.1.4 Wastage on Vessel Head Flange

Two areas of wastage were identified, between studs 23-24 and 24-25. Evaluations performed on these areas determined that the degradation was acceptable because the ligaments between studs is not a highly stressed area. The critical area on the flange is beneath the nuts and this area was not degraded.

5.1.5 Corrosion of Closure Studs, Nuts and Washers

Although visual examination revealed degradation on several studs, nuts and washers, the pressure retaining capability of the studs was not compromised at the time of plant shutdown since no wastage occurred in the load bearing portions of the studs or nuts. Multiple failure of the closure studs was therefore not a concern.

5.1.6 Cold Spot Effect on Vessel Head Fatigue Life

The possibility of a localized cold spot created by the absence of insulation due to corrosion and the air flow from the CRDM coolers was evaluated by performing a calculation in accordance with the ASME Code to determine the effect on fatigue life of the vessel head. The calculation verified that the effect of the cold spot would be well within the applicable design limit.

5.1.7 Conclusion

As a result of the above evaluation, FPL concluded that at no time was the reactor coolant system pressure boundary reduced below its design basis, nor was the operability of equipment or components required for safe shutdown degraded below their design bases as a result of the leakage.

5.2 Analysis of Postulated Continued Operation with Conoseal Leakage

In order to determine the potential consequences of the conoseal leakage if FPL had not detected and repaired the leak in Spring 1987, FPL postulated that Unit 4 continued to operate until March 1988 without repair of the leak. The results of FPL's analysis are summarized below.

FPL evaluated the various components affected by the conoseal leak to determine which component would be expected to fail first as a result of corrosion by boric acid from the leak. These components included the conoseal clamp, the reactor vessel (including the reactor vessel head and studs), the CRDM and associated components, and the vent shroud support assembly. Corrosion of the conoseal closure bolt would have resulted in leakage before significant wastage occurs on the vessel head or flange. With the vessel penetration and attachment welds being Inconel, these would not have been affected by boric acid corrosion. Based on an extrapolation of the as found inspection of the CRDMs and coolers, it is expected that neither would have been compromised so as not to perform as required during a plant shutdown. Reactor vessel stud holes would have been expected to maintain their integrity during this time. Any additional wastage on the vessel nozzles and nozzle supports would have been negligible. No significant additional corrosion of the vent shroud support assembly is anticipated which would have affected the operation of any safety related equipment. Based on these evaluations, FPL has determined that the first component which would have failed would be the conoseal clamp (and in particular, the closure bolt for the clamp).

Failure of the clamp closure bolt would have resulted in an increase in the leakage from the conoseal. Any large leakage accompanied by gross conoseal failure, if in fact it occurred at all, would have been preceded by a small detectable leak. As the leakage approached a significant level (i.e., above 1 gpm), equipment for the detection of leakage, radiation alarms, and increased coolant makeup would have alerted operators to the existence of an unidentified leak so that an evaluation would have been conducted to determine the need for shutdown.

The potential for multiple failure of the reactor vessel closure studs due to long term exposure to boric acid has been investigated. Exposure of the studs under the projected time frame until March 1988 is not expected to have progressed to the point where preload would have been lost. The condition is expected to result in the studs gradually losing their effectiveness, which would have resulted in initiation of, and then a gradual increase in, leakage through the reactor vessel flange O-ring. Such leakage would have been detected by plant

operations and the unit would have shut down. At the time of plant shutdown, it was verified that none of the studs had experienced loss of effectiveness. This indicates that at no time did the possibility for multiple failure of the reactor vessel closure studs exist, nor would such a possibility have existed had operation continued until the March 1988 outage.

6.0 ANALYSIS OF POTENTIAL LEAKAGE MECHANISMS AND CORRECTIVE ACTION

Section 4.0 describes the inspections, evaluation and corrective action for the individual components which were affected by the conoseal leak. The purpose of this section is to discuss the potential leakage mechanisms associated with the conoseal leak and identify the actions which FPL has taken or is planning to take to prevent recurrence of this type of event.

The remainder of this section is divided into three parts. Section 6.1 provides a summary of the potential leak mechanisms. Section 6.2 addresses actions which are being taken to improve leak detection, evaluation, and repairs at Turkey Point. Section 6.3 discusses other relevant action FPL is taking at Turkey Point.

6.1 Analysis of Potential Leakage Mechanisms and Actions to Prevent Recurrence

Investigations were performed to identify potential leakage mechanisms associated with the conoseal leakage. As a result of the inspections discussed in Section 4.2, it was determined that two components in the conoseal assembly had significant damage and may have been associated with the conoseal leakage mechanisms. These components are the clamp shim and the conoseal gasket.

The clamp shim was severely deteriorated and found in two pieces as a result of corrosion (the shim was not made of a corrosion resistant material). Corrosion of the clamp shim could have reduced the load imposed by the clamp assembly on the conoseal assembly. In turn, this could have influenced the ability of the assembly to function without a leak.

The original source of the corrosion of the shim could not be determined. The source may have been the conoseal leak itself; on the other hand, the source of the initial corrosion of the shim may have been an unknown external leak from another component.

The conoseal gasket had erosion to the outer seating surface. This erosion was caused by the leakage and provides the pathway for the leakage to escape from the conoseal assembly.

There are several potential mechanisms which may have resulted in the initiation of the leak and erosion of the gasket, and

it was not possible to confirm the actual mechanism. Corrosion of the shim could have influenced the leak. Additionally, debris or imperfections in the conoseal could have provided a pathway for the leakage. However, any evidence of debris or imperfections would have been eliminated by the leak itself, and QC did not identify any debris or imperfections during its inspections of the assembly of the conoseal.

Additionally, it is possible that steps in the assembly sequence of the conoseal were not properly performed, resulting in an imperfect seal. QC did not identify any inadequacies during installation. However, not every step is subjected to QC verification, so the possibility of inadequacies in installation cannot be conclusively refuted.

In order to eliminate a recurrence, FPL has decided to treat the most likely potential leakage mechanisms identified in this analysis.

The first potential mechanism would be addressed if the lower conoseal clamp and shim were made of a corrosion resistant material. FPL had decided, prior to this occurrence, to order replacement instrumentation port column seal assemblies which used a new Westinghouse design. This design uses a corrosion resistant clamp with no shims. Other benefits of the new design are:

- reduction in personnel radiation exposure for ALARA purposes
- simplified assembly effort
- reduction of assembly time

Although the design change does not require replacement of the female flange, FPL decided to replace the female flange because of the erosion that occurred.

In conjunction with the hardware changes discussed in this report, the procedures affecting the conoseal are being revised. This will address the second potential mechanism in particular. Enhanced training of installation and inspection personnel and revised installation and inspection procedures will reduce the probability of debris or imperfections on the flanges and gasket going undiscovered.

Based on experience gathered during its review of potential leakage mechanisms, FPL will implement the following prior to return to service:

- use new Westinghouse conoseal assemblies with the Articu-clamp design and provide training to installers and inspectors;

- o revise maintenance procedure to provide for enhanced QC inspections for debris, scratches, nicks, and burrs;
- o provide protection of female flange sealing surfaces while exposed;
- o revise pre-op overpressure test procedure to require visual check of the entire conoseal assembly including thermocouple conduits and swaged fittings.

6.2 Improvements in Leak Detection, Evaluation, and Repair

In order to prevent recurrence of the type of problem which developed with respect to the conoseal leakage, an evaluation of all relevant events associated with the leakage up to the point of initiating corrective measures on March 13, 1987 has been performed.

These events were then analyzed to determine the actions which might be taken to improve leak detection, evaluations, and repair at Turkey Point.

6.2.1 Leak Detection

Turkey Point is taking several additional steps to ensure that leaks in borated water systems are detected as soon as possible.

FIRST, the accessible reactor head area will be visually inspected for boron deposits and other evidence of primary system leakage whenever the plant is taken from Mode 2 to Mode 3 (prior to return to Mode 2) if an inspection has not been performed in the last 30 days.

SECOND, current requirements for leak inspections will be revised to include appropriate leak inspections during RCS filling and venting operations of components inside containment that have undergone disassembly or maintenance.

THIRD, leak inspection procedures will be revised to add more specific instructions for inspecting components that could leak on the reactor head.

Enhancement of leak detection techniques will be pursued through industry groups.

6.2.2 Disposition of Identified Leaks

Although most primary coolant leaks can drain directly to the floor of containment, leakage in some cases, such as this leak in the head area, can be retained by insulation or other structures so that the boric acid can accumulate

which can then create corrosion problems. In view of this, Turkey Point is taking several steps to ensure that such leaks are promptly repaired once detected.

First, Turkey Point will formalize its existing program which provides for inspection of leaks in containment prior to cooldown. This will ensure that leaks can be identified while systems are at operating pressure and temperature and subsequently repaired at cold shutdown. These inspections will be performed taking technical specifications and radiological conditions into account.

Second, for any leak discovered in the containment, whether or not the leak is a result of a scheduled or required inspection, Turkey Point will perform an inspection of the leak in accordance with procedures and document the inspection for subsequent evaluation as per 6.2.3.

6.2.3 Leak Evaluation

The technical specifications for Turkey Point contain limits on operating with leaks in the reactor coolant system (RCS), including a requirement to perform an evaluation if leakage exceeds 1 gpm. As an aid in implementing this requirement and in order to provide additional and more formal requirements for evaluation of leaks, FPL is revising its procedures for evaluation of leaks in the RCS.

To ensure that all potential effects of a leak are properly evaluated, Turkey Point will develop specific procedural guidance for performing an evaluation based on the results of a leak inspection. This will include criteria for determining if a given leak is acceptable, or if immediate repairs are required.

6.3 Preparation and Review of Safety Evaluations

On occasion the engineering department is requested to perform a specific safety evaluation of RCS leakage. The intent of a safety evaluation is to ensure that the plant remains within its design basis. As described in Section 3.2, FPL performed a safety evaluation of the conoseal leak in August 1986 and concluded that operation of Unit 4 with the leak for six months (until the next inspection of the leak) would be within the design basis. This conclusion proved to be valid.

However, the conoseal leak incident indicated that long term operation with RCS leakage could create conditions that, while not exceeding the design basis, could be highly undesirable. Accordingly FPL is reviewing, and will update as appropriate, its procedures concerning operation with RCS leakage as discussed in section 6.2. FPL will also review and update, as appropriate, its guidelines concerning evaluations with particular attention to consideration of secondary effects. These guidelines will be utilized during the preparation and review of safety evaluations for RCS leakage.

7.0 CONCLUSION

In summary, FPL is:

1) Performing inspections to identify the extent of the boric acid deposition resulting from the conoseal leakage.
2) Evaluating the affected components to determine whether they are acceptable for use.
3) Replacing, repairing or cleaning components as warranted, and
4) Taking action to prevent recurrence of events similar to the conoseal leakage.

As stated in Safety Analysis Section, FPL has concluded that at no time was the reactor coolant system pressure boundary reduced below its design basis, nor was the operability of equipment or components required for safe shutdown degraded below their design basis as a result of the leakage.

Upon completion of the actions associated with preparation for startup which are discussed in Sections 4 and 6, it is FPL's conclusion that the issues associated with this event will have been resolved and would not prevent return to normal operation.