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AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Operating Reactors

SUBJECT: Forwards "Steam Generator Repair Rept," Revision 7 re lower assembly removal,dose calculations,steam generator disposal & return to svc testing.

Revised 5/24/80 by Joanne M.

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FLORIDA POWER & LIGHT COMPANY

March 28, 1980
L-80-108

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut
Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Steam Generator Repair Report

Attached herewith is Revision 7 to Florida Power & Light Company's Steam Generator Repair Report (SGRR) which was forwarded to you on September 20, 1977 (L-77-296).

The major changes included in Revision 7 are summarized below.

(1) Lower Assembly Removal

The SGRR, through Revision 6, has described the repair as involving removal of the upper assembly by cutting the steam generator in the transition cone area and removing the lower assembly by cutting the reactor coolant piping. This method is referred to as the pipe cut approach. However, due to recent Surry experience, we have re-evaluated the methodology and man-rem exposures associated with the pipe cut approach and have determined that an alternate approach would be more desirable. This revision revises the SGRR to reflect this new approach. It consists of removing the upper assembly, as before, by cutting in the transition cone area. Removal of the lower assembly would be accomplished by cutting the lower assembly near the junction of the steam generator channel head to the tubesheet. This alternative is referred to as the channel cut approach. This approach is expected to result in a significant savings of man-rem over the pipe cut approach.

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(2) Dose Calculations

As part of our re-evaluations, in addition to revising man-rem dose estimates, we have also revised off-site

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dose calculations to reflect the non-uniform deposition of corrosion products on the various internal surfaces of the steam generators. Previous release calculations has assumed uniform deposition. The relative concentration factors used in these revised calculations are shown in Table 6.2-1.

We have also updated the SGRR to reflect actual 1978-1979 releases and exposure data. This data is used throughout the report for comparison purposes with the releases and exposures expected from the repair. Previously, 1976 data had been used.

(3) Steam Generator Disposal

As part of our on-going evaluations, we have now determined that disposing of the old steam generators by shipping them by barge to a land burial site may now be a viable alternative. Barge shipment had previously not been considered viable because of the lack of barge-slip facilities at the Barnwell, S. C. disposal site. We now feel that the appropriate facilities can be made available at Barnwell. Chapters 3 and 8 have been revised to reflect this.

(4) Return To Service Testing

Chapter 4 of the SGRR has been revised to include additional details on the return to service testing which will be done to verify the integrity of the reactor coolant system and the containment building. This information is in response to the Atomic Safety and Licensing Board order dated October 11, 1979.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/GDW/ah

Attachment

cc: J. P. O'Reilly, Region II
Harold F. Reis, Esquire

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E. Uhrig

Robert E. Uhrig

Subscribed and sworn to before me this

28th day of March, 1980

Lois T. Marino

NOTARY PUBLIC, in and for the county of Dade, State of Florida

My commission expires: NOTARY PUBLIC STATE OF FLORIDA at LARGE MY COMMISSION EXPIRES AUGUST 24, 1981 BONDED THRU MAYNARD BONDING AGENCY

DIRECTIONS FOR INSERTING REVISION 7, MARCH 1980

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50-250 Revision 7 to Steam Generator Repair Rpt 3/28/80

#8004010210 Joanne Morrison

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STEAM GENERATOR REPAIR REPORT

TURKEY POINT UNITS 3 AND 4

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STEAM GENERATOR REPAIR REPORT
TURKEY POINT UNITS 3 AND 4

1.0 INTRODUCTION, SUMMARY AND CONCLUSIONS

The steam generators at Turkey Point Units 3 and 4 have experienced corrosion related phenomena that require periodic inspection and plugging of steam generator tubes to ensure their continued safe operation. It is conceivable that continuation of the corrosion mechanism, referred to as denting, could eventually result in unacceptable inspection intervals and reductions in unit power. Thus, primarily economic considerations may require repair of the steam generators.

The Florida Power & Light Company is currently pursuing engineering and
licensing activities to enable it to affect a timely steam generator repair 1
on Unit 3 and/or Unit 4 should the need arise.

This report discusses the safety-related aspects associated with repair of the steam generators via replacement of the lower portion of the in situ units with shop fabricated replacement lower assemblies. It does not address the in situ retubing alternate.

1.1 SUMMARY OF STEAM GENERATOR REPAIR EVALUATION PROGRAM

1.1.1 Containment Entry and Exit of Steam Generator Lower Assemblies

A 1/2" - 1'0" scale model of the Unit 4 containment and internals and the Unit 3 containment in the area of the equipment hatch has been constructed to evaluate the feasibility of various pathways for removal of lower assemblies from containment and return of the shop fabricated lower assemblies to containment. Three pathways were evaluated in detail, viz., removal and entry via:

a. the equipment hatch,

b. an opening in the containment dome and

c. an opening in the containment wall.

Schedule, economic and safety evaluations indicated that use of the equipment hatch is the optimum pathway. Therefore, construction-related evaluations addressed in this report are for the equipment hatch pathway only.

1.1.2 Steam Generator Lower Assembly Characteristics

The existing steam generators will be parted in the upper section of the shell. The steam dome assemblies (upper portion of steam generator) will be removed and stored within containment. Subsequent to completion of the installation of the new lower assemblies, the original steam dome assemblies will be welded to the new lower assemblies to complete the repair.

The shop-fabricated lower assemblies (see Figure 2.2-1) will be equivalent to the lower assemblies they replace. They will be designed to meet existing plant mechanical and performance characteristics, and safety-related parameters will remain consistent with those utilized in the FSAR analyses.

Features to mitigate the effects of corrosion-related phenomena will be incorporated in the design. These features will not adversely alter mechanical, performance or FSAR-related characteristics. In addition, the shop-fabricated lower assemblies will be designed and manufactured to current codes and manufacturing techniques. Thus, the replacement assemblies will reflect current technology. They will satisfy the legal requirement of being equivalent to the units they replace (which were manufactured to the 1965 edition of Section III, ASME Boiler and Pressure Vessel Code).

1.1.3 Safety-Related Considerations

The potential impact of the repaired units on each appropriate accident analyzed in the FSAR has been evaluated. Because of the essential duplication of safety-related parameters, qualitative discussion is sufficient to demonstrate the appropriateness of the repaired steam generators to accommodate FSAR accidents.

Onsite transportation and handling of lower assemblies have been evaluated. Even though administrative procedures and temporary protection to buried pipes and conduits are considered sufficient to preclude occurrence of these types of mishaps, construction incidents are postulated to occur. The ability of the plant to accommodate these events is demonstrated.

The units will be repaired in series, i.e., one unit will be conducting normal power operations while the second is undergoing steam generator repair. To obviate the need to evaluate construction incidents within the containment undergoing repair, the reactor core will be off-loaded and transferred to the fuel storage building prior to commencement of major repair activities within 1
containment.

1.1.4 ALARA Considerations

Estimates have been made of the exposure to personnel involved in the repair activity. This evaluation indicates that the reduction in man-rem exposures currently being incurred during tube inspection and plugging operations will offset, in a reasonable time period, the man-rem exposure incurred during the steam generator repair.

1.1.5 Offsite Radiological Considerations

Radiological evaluations of the gaseous and liquid releases attributable to the steam generator repair have been conducted. The resulting releases are a small fraction of the total releases in 1976, i.e., those associated with normal operation of the facility. The steam generator repair will in effect provide a reduction in future radioactive effluents from the plant because the repair will result in enhanced steam generator integrity, thus reducing primary to secondary system leakage. 1

1.1.6 Unique Aspects of the Program

The shop fabrication of the lower assemblies will be conducted in accordance with standard practices. Welding of the steam dome assembly to the lower assembly in the field was utilized in the installation of the existing steam generators, which were shipped in two sections. Concrete removal and replacement will be accomplished utilizing standard construction practices. Transport and lifts of heavy vessels well in excess of the approximately 205-ton lower assemblies are commonplace. Movement of heavy loads across buried pipe and electrical conduit is commonplace during construction of power plants, e.g., during construction of Units 3 and 4 the steam generators and reactor vessel were transported over the same pipe and conduit that the lower assemblies will pass over, without incident. In summary, the repair program will utilize tried and proven manufacturing and construction practices.

1.1.7 Steam Generator Disposal

The repair activity and ultimate disposal of the existing lower assemblies are separable issues. This report discusses the various means by which the steam generators can be disposed of to demonstrate the feasibility of disposal. The evaluation indicates that economic considerations will likely determine the method to be utilized. During the time between removal from containment and ultimate disposal, the lower assemblies will be stored onsite in a temporary storage facility.

1.2 IDENTIFICATION OF PRINCIPAL AGENTS AND CONTRACTORS

The Florida Power & Light Company, hereinafter FPL, is a public utility corporation duly authorized and existing under the laws of the state of Florida. FPL is the sole owner and operator of the Turkey Point Plant.

FPL has been actively engaged in nuclear power for a number of years and currently operates 3 nuclear power units: Turkey Point Units 3 and 4 and St. Lucie Unit 1. This represents a total operating experience of approximately 11 reactor years. Construction of St. Lucie Unit 2 is in progress. FPL will have overall responsibility for the steam generator repair program.

Bechtel Power Corporation has been retained by FPL as the Architect Engineer. Bechtel has been continuously engaged in construction and engineering activities since 1898. Since the close of World War II, Bechtel has placed strong emphasis on electrical power generation projects. During this period, Bechtel has been responsible for the design and/or construction of more than 237 thermal generating units, representing more than 118,000 MW_e of new generating capacity. Of this number, a nuclear capacity of more than 69,000 MW_e has been or is currently being engineered.

For more than 20 years, Bechtel has been actively working on nuclear power plants. Responsibiliies have covered design, construction, startup, site surveys, license applications, feasibility studies, and equipment procurement. The ratings of thermal generating plants designed by Bechtel range to 1,300 MW_e per unit and include most types of station design and arrangement. The majority of contracts for these facilities provided Bechtel with complete responsibility for engineering, construction, and startup.

3.0 COMPONENT REPLACEMENT PROGRAM AND PROCEDURES

This section discusses the engineering evaluation of the field activities required to implement the steam generator repair. Figure 3.0-1, Outage Sequence, and Figure 3.0-2, Removal Sequence, illustrate the lower assembly removal sequence. It should be noted that implementation methods and procedures may vary from that described below as engineering is finalized. The methods below are provided to demonstrate feasibility of implementation. Any changes incurred during detailed design will not alter the envelope of construction incidents postulated in Section 5.2.

The steam generator lower assemblies will be removed and replaced through the equipment hatch. The steam dome assemblies will remain inside the containment and will likely be stored at the operating floor level, elevation +58', until the new lower assemblies are brought in and installed.

Handling of the steam generator lower assemblies inside the containment will require an additional polar crane trolley with a capacity in the range of 250 tons. (Refer to Figure 3.0-2 sequences I through VI.) Handling through the hatch will require special track-mounted tailing devices attached to the steam generator vertical support lugs. (Refer to Figure 3.0-2 sequences VII through XI.) Handling outside the containment will likely be by crawler lift crane.

A rigging platform will be provided inside the containment. This platform will provide support for the lower assemblies while they are being maneuvered from the vertical position to the horizontal position. The temporary construction loads from this platform will be transmitted to the containment base mat. In addition, temporary laydown area will be provided for three steam dome assemblies weighing 100 tons each and for three sets of swirl vane assemblies weighing approximately 5 tons each. Adequate laydown area is available inside the containment, utilizing in some cases temporary beams to span certain operating floor slab areas.

Clearances to accomplish the removal of the lower assembly through the equipment hatch will be provided through the removal of internal concrete in the vicinity of the equipment hatch. The proposed portions to be removed are listed in Section 3.2.5. Impact on existing equipment is minimal and is described in Section 3.2.

The removal of lower assemblies through the existing equipment hatch will have minimal impact on the site layout in terms of new foundations required. Upon exiting from the equipment hatch, the lower assemblies will likely be lifted onto a lowboy type trailer or rubber wheeled transporter by one or more large cranes and moved to a suitable location for temporary onsite storage. Special foundations will be required only for the access platform described below.

The existing access ramp to the equipment hatch will be removed as necessary to enlarge the access area and to facilitate access by trucks and mobile cranes. A platform approximately 30' x 30' will likely be installed at the equipment hatch elevation.

Removal of the existing ramp and grading of the general area will provide additional flexibility in selection of cranes for the heavy lifts during steam generator movement.

No permanent modifications to existing structures are expected. | 1

3.1 PATHWAYS AND CONSTRUCTION RESTRICTIONS

3.1.1 Site Preparation

3.1.1.1 Foundations

All heavy hoisting equipment will be located so that foundations will not interfere with permanent plant installations, either above or below grade. The steam generator removal platform will require the placement of foundations as shown in Figure 3.1-1. Below grade emplacements may be removed after completion of the repair work.

3.1.1.2 Roadways, Ramps, and Platforms

The existing access ramps to the equipment hatch of each containment will be removed as necessary. A service platform (Figure 3.1-1) may be installed to facilitate access into the containment. The platform will be designed to support the weight of the lower assemblies and a manually powered materials cart of approximately 20-ton capacity. It will also serve as a loading dock for the loading/unloading of a number of trucks or mobile cranes simultaneously. The platform will be equipped with jib cranes of 2 to 5 ton capacity for this purpose.

Onsite steam generator haul routes from the Unit 3 & 4 barge slip, i.e., the slip used during construction of the facility, are shown in Figure 3.1-2. Actual haul routes may vary during detailed engineering depending on method | 1 of receipt of lower assemblies and location of temporary onsite disposal area. This notwithstanding, all haul routes for transportation of heavy loads (other than normal axle loads for highway equipment) will be evaluated and upgraded where necessary utilizing standard construction practices (see Subsection 3.1.1.3).

As required, additional ramps to access plant grade will be designed for the maximum wheel loads of anticipated transportation equipment. | 1

3.1.1.3 Protection of Buried Facilities

Analyses have been performed for the following safety-related Units 3 and 4 buried facilities to ascertain in situ capability to accommodate transport loads (refer to Figure 3.1-2 for locations). These safety-related buried facilities include all those potentially affected by the haul routes associated with each steam generator site receipt alternative. | 1

a. Electrical duct banks beneath the onsite steam generator haul route 1

b. Intake cooling water piping beneath the onsite steam generator haul route 1

c. Electrical duct banks beneath the equipment hatch construction area for Unit 3

d. Intake cooling water piping beneath the equipment hatch construction area for Unit 4

The following handling and transporter equipment was assumed for these analyses:

a. A multi-axle trailer/dolly system would be used to transport the lower steam generator assembly along the onsite haul routes. The total transporter load assumed was 275 tons.

b. One 300-ton capacity crawler crane weighing approximately 245 tons, including crane, a 70-foot crane boom, and counter weight, would most likely be used in the equipment hatch construction area.

The actual external earth pressure due to earth cover and equipment surcharge was determined for comparison with the original Turkey Point design allowables. The results for buried piping, presented in Table 3.1-1, do not include the use of methods such as additional structural fill, timber matting or structural bridging to distribute surcharge loads. Based upon the results of these analyses, additional protection will not be required for the safety-related intake cooling water pipe.

All buried concrete electrical duct banks were analyzed as infinite unreinforced concrete beams on an elastic foundation. These analyses indicate that uniform distribution of the equipment surcharge loads by the use of timber matting or structural fill will be required to maintain concrete tensile stresses within allowable limits. Therefore, the results of the analysis for buried duct banks, presented in Table 3.1-2, are based upon typical timber matting. The duct banks under the haul route are actually reinforced and will have a higher factor of safety than that presented in Table 3.1-2. 1

Timber matting or the equivalent will be required under crane outriggers.

In all cases, use of timber matting or other equivalent protection will ensure that allowable stresses in buried duct banks and pipes are not exceeded. 1

3.1.1.4 Steam Generator Receipt on Site

The method of transportation to the site will be either by ocean barge or overland from the nearest railhead.

3.1.1.5 Laydown Facilities Outside Containment

Adequate laydown area for construction materials and equipment is presently available. This area is rough graded fill material placed over subsurface muck to a final elevation of +5'.

This area is adequate for its intended use without further improvement. Access to adjacent plant areas will be obtained by construction of ramps and roads as mentioned in Subsection 3.1.1.2.

3.1.2 Containment Preparation

3.1.2.1 Temporary Platform Inside Containment

A temporary platform at elevation +30'-6" inside the equipment hatch opening will be provided to handle the steam generators (Figure 3.1-1). The existing deck at this elevation will be removed. Guide rails on this structure will extend through the hatch opening and align with similar guide rails on the outside platform, providing a path for the roller assemblies required to permit steam generator movement through the hatch.

3.1.2.2 Other Preparations

Concrete removal and equipment relocation for rigging clearances are discussed in Section 3.2.

3.1.2.3 Laydown Facilities Inside Containment

All fuel assemblies will be removed from the reactor and stored in the spent fuel pool, the reactor internals stored in the reactor, the reactor head stored in place on the reactor, and the control rod drive mechanism (CRDM) missile shield stored on top of the refueling cavity walls. In addition, the water will be drained from the refueling cavity. Laydown of each of the steam dome assemblies may be as follows:

a. One on the CRDM missile shield (Figure 3.1-3)

b. One on elevation +58' over an RCP motor hatch (Figure 3.1-4)

c. One in the reactor head laydown area (Figure 3.1-5)

There is adequate space inside containment for the temporary storage of the additional equipment discussed in Section 3.2.

3.1.2.4 Containment Structural Analyses

Containment structural analyses have been performed in accordance with the design criteria in Appendix 5B of the Turkey Point FSAR for the following:

a. Temporary laydown areas at elevation +58'-0" on the operating floor. These areas will be required to support the steam dome assemblies, pipe sections, and miscellaneous construction equipment. (See Figures 3.1-3 through 3.1-5.)

b. Containment base mat and existing floor support embeds in the containment wall. A temporary transfer bridge platform will be installed inside and outside the containment to facilitate the removal of the lower assemblies. Platform loads inside the containment will be transmitted to the base mat. Platform loads outside the containment will be transmitted to structural fill. (See Figure 3.1-1.)

These analyses indicate that the containment, foundation and internal structures, including the CRDM missile shield, are capable of supporting the construction loads without permanent modifications to the existing structures.

3.1.3 Transportation On-Site

Movement of the new steam generator lower assemblies (205 tons) on site can be accomplished by several methods such as drop-deck or flatdeck trailer. Motive power may be rubber-tired tractor or tracked vehicle, as required.

3.1.4 Rigging Configuration

3.1.4.1 Inside Containment

The existing polar crane bridge is structurally adequate to sustain the loads imposed by a lower assembly in addition to a 250-ton capacity construction hoist weighing approximately 20 tons. Administrative controls will ensure that design bridge truck wheel loading is not exceeded during rotation by requiring that a steam generator lower assembly be moved toward the center of the bridge prior to bridge rotation.

Because of insufficient lift capacity, the existing polar crane trolley is not suitable for steam generator lower assembly replacement, and will, therefore, be moved aside to permit the placement of the 250-ton construction hoist on the polar crane bridge. The temporary hoist will be load tested to meet current OSHA Safety Standards prior to its use for construction lifts.

The steam dome assemblies will be parted from the lower assemblies and lifted by existing pad eyes and commercial sling assemblies and relocated to selected storage locations, as discussed in Subsection 3.1.2.3.

The lower assemblies will be lifted from their compartments using conventional hoisting techniques. The hoist lower load block will be linked by pins to a steam generator lift beam equipped with toggle arms or endless grommet type cables. The toggle arms or cables will engage existing lifting trunnions on the assemblies. Each lower assembly will be lifted and transferred in turn to a point approximately 12 feet from the containment inside wall and approximately on the centerline of the equipment hatch. Special tilting assemblies,

such as Hillman roller units and structural members, will be required to permit the assembly to move from the vertical to the horizontal position. Transfer of the lower assemblies through the equipment hatch will require the connection of additional roller assemblies as each lower assembly travels beyond the reach of the polar crane hoist.

3.1.4.2 Outside Containment

The lower assembly will exit the containment approximately 14 feet above grade, on the access platform previously described in Subsection 3.1.1.2. Transfer to a trailer/dolly system will be accomplished by a suitable lifting device.

Shipping saddles and tie downs will be provided for secure attachment while the transport device is in transit to the storage/laydown area.

3.1.5 Rigging and Handling Controls

All lift cranes and transport devices will be located so that postulated boom pendant failure will result in a path of travel that does not adversely impact the ability to achieve and maintain safe-shutdown conditions and provide adequate cooling for stored spent fuel. In order to develop the required rigging and handling controls, it is postulated that a crane boom failure will occur in the plane of normal travel of the boom, i.e., either forward, or as a reaction rearward over the cab of the crane. Administrative controls will limit lift heights so that loads will be raised only to a height sufficient to provide clearance for horizontal movement.

Although administrative controls will be provided, subsection 5.2.1.2 discusses the analyses of postulated crane boom failures. Based on this evaluation, all structures evaluated will maintain structural integrity for the postulated 1
crane boom failure. When traversing the east plant road in the vicinity of the nuclear units, crane booms will be in the lowered position.

Travel speed and travel routes for cranes and transport devices will be controlled to minimize their influence on vital structures in the immediate area. A typical lower assembly haul route is shown in Figure 3.1-2. Controls 1
on the rotation of the polar crane when loaded are discussed in Subsection 3.1.4.1.

3.2 EQUIPMENT AND CONCRETE REMOVAL AND REPLACEMENT

Engineering evaluations, including model sequence studies, have been conducted to determine the impact of repair activities on equipment and structures in containment. This evaluation demonstrates that the repair activity will not result in any safety considerations due to equipment removal or interruption of function. It also demonstrates that there will not be any major impact on structures and equipment (non-steam generator related).

Detailed engineering studies are in progress to precisely define the components, pipes, cables, instruments, etc. within the containment affected by the repair activity. The discussion below provides the results of the study to date. It is provided to illustrate the minimal impact on non-steam generator related equipment within containment.

3.2.1 Mechanical Equipment

The following equipment which interferes with the lower assembly pathway will be temporarily removed and relocated inside the containment:

a. The fan and diffuser assemblies will be unbolted and removed from the control rod drive coolers.

b. The containment emergency air coolers (3V30A and 4V30C) nearest the containment equipment hatch opening will be unbolted, disconnected from their cooling water piping and moved clear of the pathway.

c. The Unit 3 reactor coolant pump motor 3P200A opposite the equipment hatch will be disconnected from its pump and cooling water piping and moved clear of the pathway.

As appropriate, equipment within the containment will be covered to ensure cleanliness during the repair.

Upon completion of the repair, affected equipment will be returned to service using standard procedures followed during routine plant maintenance programs.

Disconnection of power cables to the above equipment is discussed in Section 3.2.3.

3.2.2 Instrumentation

The following instrumentation, sensing lines, and associated supports will be temporarily removed and relocated inside the containment:

a. Unit 4 steam generator "C" level transmitters LT-494 and 497 and supports which are mounted to the removable secondary shield wall panels opposite the equipment hatch

b. The high and low pressure sensing lines for the level transmitters discussed in a. above

c. The high and low pressure sensing lines located on the shield walls above elevation + 58' for the remaining Unit 4 steam generator "A", "B" and "C" level transmitters

d. The sensing line support structures for the sensing lines discussed in b. and c. above

e. Unit 4 ECCS accumulator 4T-229C, pressure transmitter PT4-931 and level transmitter LT4-930

f. The associated sensing lines for the transmitters discussed in e. above

The open ends of lines will be capped to ensure cleanliness during the repair.

As appropriate, the instrumentation and sensing lines will be returned to service using standard procedures followed during routine plant maintenance programs.

Disconnection of instrumentation cables to the above transmitters is discussed in Section 3.2.3.

3.2.3 Cable and Conduit

The steam generator repair does not require the removal and relocation of major pieces of electrical and control equipment such as panels, load centers, transformers or motor control centers. Only power and instrumentation cable and conduit as described below are affected.

Moving the control rod drive cooler fan and the containment emergency air cooler discussed in Section 3.2.1. will necessitate disconnecting the fan motor power cables. The cable terminations will be disconnected and the cables pulled back and coiled out of the path of the lower assembly. The same cable will then be reconnected when the fans are returned to their original location.

Table 3.2-1 is provided to illustrate Unit 3 & 4 power and instrumentation 5
conduit (including the instruments discussed in Section 3.2.2), to be temporarily removed.

The conduit to be removed will be tagged, disassembled, and the associated cable pulled back and coiled. When the conduit is later reinstalled, the cable will then be repulled and reconnected. Procedures will be generated for pulling back, coiling and repulling of cables and removal and reinstallation of the conduit. Circuit checkout procedures will also be written.

3.2.4 Piping

In order to accomplish the steam generator repair it will be necessary to cut portions of the following piping systems:

a. Reactor coolant piping

b. Main steam piping, including small pipe vent lines

c. Main feedwater piping

d. Steam generator blowdown piping

e. Cooling water return piping from the containment emergency air cooler

f. Cooling water supply and return piping to reactor coolant pump motor "A" for Unit 3 only

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g. Service air piping

h. Primary service water piping

Location of cut areas for reactor coolant system, main steam system, and main feedwater system piping are shown on Figures 3.2-1, 3.2-2, and 3.2-3, respectively. As appropriate the open ends of cut piping will be covered to ensure cleanliness during the repair.

Piping weld end preps, welding and nondestructive examination for the reinstallation will be in accordance with the latest edition of the ASME Boiler and Pressure Vessel Code, Sections III, V, IX and XI. The piping system will be reinstalled in accordance with FSAR criteria.

3.2.5 Concrete and Structural Steel

The following structures or portions of structures within the containment will be removed to provide a path for the lower assembly (refer to Figures 3.2-4 through 3.2-6 for illustration):

a. A section of the steam generator "C" shield wall above elevation + 58' for Unit 4. A section of the steam generator "A" shield wall above elevating + 58' for Unit 3.

b. A section of the operating floor concrete at elevation + 58' including a steam generator "A" upper support embed for Unit 3 and steam generator "C" thrust beam for Unit 4

c. The removable secondary shield wall panels opposite the containment equipment hatch from elevation + 30'-6" to elevation + 58', and an additional width of secondary shield wall opposite the equipment hatch from elevation + 30'-6" to elevation + 58'

d. A portion of the floor framing and grating at elevation + 58' above the equipment hatch

e. A portion of the floor framing and removable floor slabs at elevation + 30'-6" at the low point of the equipment hatch

f. The upper portion of the steel stairway near the equipment hatch opening

g. A reinforced grouted pad in the equipment hatch at elevation + 30'-6"

h. A portion of the truss system tie rods to allow for clearance of the temporary polar crane trolley. (The truss system was originally utilized in the construction of the containment and does not perform any structural related function at present.)

3.2.5.1 Removal of Concrete Structures

Removal of bulk volumes of containment internal structural concrete will utilize equipment and techniques commercially available. The intent is

3.3.3 Control of Airborne Radioactivity and Surface Contamination

Airborne radioactivity inside containment during the steam generator repair effort will be controlled, monitored and ultimately released via the plant vent stack. A slightly negative pressure inside of containment will be maintained using the containment purge exhaust system. Air will be drawn through the equipment, personnel and emergency hatches and exhausted by the purge system via the plant vent, thus precluding airborne radioactive particles or gases from leaving containment openings utilized for construction activities. Air flow requirements necessary for maintaining a slightly negative containment pressure are well within the existing purge exhaust system capacity. The air being exhausted will be monitored as it passes the 1
existing sampling station located within the main plant vent.

In addition to bulk containment atmosphere control of airborne radioactivity, appropriate localized control will also be provided. Radioactivity generated during the cutting of the reactor coolant pipes will be contained within specially designed contamination control envelopes, which will provide local high efficiency filtration. Personnel working inside these control envelopes will wear respiratory protection equipment, as required, described and implemented by FPL procedures HP-60 through HP-69, in accordance with plant Technical Specifications and 10 CFR 20.103.

No special provisions are required for other cutting operations inside containment.

Section 3.3.1 describes the method of controlling the spread of surface contamination by personnel removing their outer set of protective clothing when leaving the control envelope.

The radioactive release and dose assessment associated with cutting the reactor coolant loop is provided in Subsections 5.2.2.1 and 5.2.2.2.

3.3.4 Supplemental Personnel Monitoring Requirements

3.3.4.1 Monitoring of Airborne Radioactivity

Mobile air monitors will be used, as required, to monitor the airborne radioactivity inside the control envelopes and in other work areas inside containment. Airborne radioactivity samplers coupled with laboratory analyses will also be employed.

3.3.4.2 Monitoring of Workers for Ingested Radioactivity

All workers who are planned to enter a high radiation area will be given an initial whole body count or bioassay at the start of their employment. Subsequently, workers will be given whole body counts or bioassays, as necessary, to comply with requirements set forth in the FPL Health Physics Manual and FPL procedure HP-31.

3.3.4.3 Personnel Monitoring

All personnel entering the radiation controlled area will be provided with personnel dosimetry TLDs in accordance with the FPL Health Physics Manual and FPL procedure HP-30.

of manpower in high radiation areas, and to reduce overall man-rem doses by minimizing the time spent in these high radiation areas.

3.3.5.2 Temporary Shielding

Shielding will be used, as necessary, to reduce the dose rates from other components such as the regenerative heat exchanger, RHR system valves, and temporary storage areas for contaminated pieces of pipe, rags, and tools. Temporary shielding will be used, as necessary, for the steam generator while it is being cut out of the reactor coolant loop and moved out of the containment. The steam generator shell will also help shield the more contaminated parts of the steam generators.

3.3.5.3 Local Decontamination

Local decontamination of the steam generators and reactor coolant piping may be performed. Decontamination of the work areas will be performed periodically depending on the contamination levels. Paper and plastic sheeting will be used to facilitate collection and cleanup of contamination. In all cases the FPL Health Physics Manual and FPL procedures HP-70 and 71 will be followed.

3.3.5.4 Low Background Radiation Waiting Areas

Low background radiation waiting areas will be established where workers must wait between tasks. Special signs will be posted to designate these areas. Signs will also be posted in high background radiation areas to warn personnel.

Health Physics will work with the job supervisors to assure that personnel not required in the work area remain in the waiting area.

3.3.5.5 Training of Craft Personnel

As appropriate, craft personnel will be given the comprehensive course in radiological protection described in the FPL Health Physics Manual. This course will consist of approximately 20 hours of instruction and demonstrations covering in detail the basic theory and practice of radiation protection principles, emergency planning, and the Radiological Protection Program. Successful completion of this course and an associated comprehensive examination will allow personnel to have unescorted access to the radiation controlled area.

Craft personnel unable to pass the examination and/or those who take only the orientation course, which consists of one 3-hour class to acquaint individuals with basic safe health physics practices and emergency procedures, will be required to be escorted in the radiation controlled area.

3.3.6 Miscellaneous Waste Disposal

3.3.6.1 Concrete Disposal

Approximately 60 cubic yards of concrete per unit will be removed from the containment internal walls and floors and will be disposed of. The majority of this concrete has an insignificant amount of transferable contamination (transferable contamination is considered insignificant if it is less than 2200 dpm/100 cm^2 per 49 CFR 173.397) without surface decontamination. The concrete which is considered contaminated, (i) may be decontaminated prior to cutting by vacuuming and/or scrubbing with detergent and water to reduce the amount of transferable contamination to as low as is reasonably achievable or, (ii) appropriately packaged for shipment. Following removal from the containment, the concrete will be shipped as "low specific activity" (LSA) material to a licensed land burial site.

3.3.6.2 Miscellaneous Dry Waste Disposal

Metal shavings from the various cutting operations and miscellaneous dry waste, such as paper, rags, etc., will be put in standard shipping containers and shipped as LSA material to a licensed land burial site.

3.3.6.3 Liquid Radwaste Disposal

There are three potential sources of radioactive liquid to be disposed of. These sources are:

a. Water drained from the reactor coolant system

b. Laundry waste water

c. Local decontamination waste fluids

The radioactive releases associated with these sources are discussed in Subsection 5.2.2.4

The reactor coolant will be processed by the chemical and volume control system, as described in Section 9.2 of the Turkey Point FSAR.

The laundry waste water may be discharged without processing due to the low activity level as indicated by the estimated laundry waste water specific activities given in Table 5.2-5. If sampling indicates processing is required, the laundry waste water will be treated as part of the normal liquid radwaste processing scheme.

The small amount of liquid waste generated as a result of local decontamination will be treated as part of the normal liquid radwaste processing scheme.

3.3.7 Man-Rem Assessment

3.3.7.1 Man-Rem Assessment for Continuing Operation

A considerable amount of radiation exposure to personnel is associated with inspecting and plugging steam generator tubes. The exposure to |1

personnel over the remaining operating lifetime of the plants can be reduced by the repair effort. This reduction is consistent with the guidance provided by Regulatory Guide 8.8 and ALARA.

The amount of radiation exposure at Turkey Point Units 3 and 4 for 1976 was 1408 man-rem of which approximately 600 man-rem was attributable to steam generator tube plugging, eddy current testing, and other inspection related activities. The eddy current testing and tube plugging techniques employed at Turkey Point are state-of-the-art; therefore, the man-rem attributed to these maintenance/inspection activities is ALARA. Current methods (1976 to date) employ a single position eddy current fixture and explosive plugging.

Eddy current testing techniques have been continually improved in terms of man-rem exposures. Attainment of the present level of design has taken several years with many revisions and improvements to the system. The current tool for conducting eddy current examinations is a single-position fixture. The technique requires only one entry for installation and one entry for removal on each side of the channel head; hence, a total of four entries are required to complete a steam generator inspection. The previously used 1
orthogonal fixture technique required one pre-installation entry, four entries for installation/repositioning, and one entry for removal on each side of the channel head; hence twelve entries were required with the orthogonal fixture technique to complete a steam generator inspection. In addition to the higher number of entries, the orthogonal fixture technique also requires more personnel support activities at the channel head manways than the single fixture technique. These considerations, coupled with the higher mechanical reliability of the single fixture technique, result in an estimated 60 to 75 percent reduction in eddy current testing personnel exposures by using the state-of-the-art single fixture technique at Turkey Point instead of the orthogonal fixture technique.

Explosive plugging of steam generator tubes was developed in the early 1970s and continues to be the most economical approach to plugging in relation to total exposure. Hand welding of plugs in a steam generator would require approximately 15 minutes of steam generator occupancy for each plug, whereas, three explosive plugs can be set and fired with less than 1 minute occupancy time. Hence, if three tubes are to be plugged, this would result in approximately 1/45 the total radiation exposure for an equivalent conventional (manually-welded) plugging operation.

Assuming that the replacement steam generator tubes maintain their integrity during the remaining operating lifetime of the plants, radiation exposure attributed to steam generator work will be reduced. It is not expected to exceed approximately 100 man-rem per year for a tube inspection and plugging 1
operation in accordance with Regulatory Guide 1.83. (This is considered to be a conservative estimate, since it is expected that these inspections will not necessitate tube plugging.) Therefore, approximately 500 man-rem will be saved each year following steam generator repair of both units. 1

3.3.7.2 Man-Rem Assessment for the Repair Effort

In order to assess the man-rem associated with the repair, detailed surveys were made of the radiation levels at the exterior of the steam generators

2. The effectiveness of temporary shielding, the time required to place such shielding and the type of remote cutting and welding techniques utilized are not precisely defined prior to completion of detailed engineering.

3. The radiation fields used in the man-rem predictions are task average. The exact radiation fields to which an individual is exposed during the work effort cannot be precisely determined, because workers associated with a specific task may not only move around within the task area, where radia- 2
tion fields could vary significantly with location relative to the sources of radiation, but may also move around outside of the task area to perform task-supportive work.

Supporting calculations have been performed to indicate the approximate man-rem expected for each task during the repair, as discussed below, and during ultimate disposition, as discussed in Section 3.4.4. The range for the total effort represents the best judgment with respect to the predicted worker dose, considering the uncertainties noted above. The variability in ranges reflects equivalent levels of confidence in the predictions. Total man-rem are not expected to exceed the upper limit of the ranges for repair and disposition of the existing steam generators.

The tasks involved in the removal and installation of the lower assemblies were considered in the man-rem assessments presented below, including:

a. Installation of scaffolding

b. Removal and replacement of insulation 1

c. Local decontamination

d. Installation of temporary supports/rigging for upper and lower assemblies

e. Preparation and cutting of steam generator transition pieces

f. Preparation and cutting of reactor coolant piping

g. Removal and storage of the steam dome assemblies

h. Sealing openings in the lower assemblies

i. Removal of portions of the steam generator shield wall adjacent to the equipment hatch

j. Removal of the lower assemblies to the temporary storage facility

k. Installation of the new steam generator lower assemblies

Based on the information presented above, it is estimated that approximately 300 to 700 man-rem per unit could be incurred during the removal of the steam generator lower assemblies and approximately 350 to 750 man-rem per unit during the installation of the new lower assemblies and other miscellaneous repair activities. The details of these estimates are given in Table 3.3-2.

Worker exposure to airborne radioactivity was also considered in the overall evaluation of radiation exposure for the repair effort. During the repair effort, the normal contributors to airborne activity inside containment, the reactor coolant system and the refueling cavity, will provide no contribution since both will be drained of radioactive liquids. In addition, as described in Section 3.3.3, airborne radioactivity inside containment will be controlled by operating the containment purge system and radioactivity generated during the cutting of the reactor coolant pipes will be contained within specially designed contamination control envelopes, which will provide high efficiency filtration. Therefore, the airborne radioactivity concentrations to which workers will be exposed for most of the tasks performed during the repair are expected to be insignificant. Two specific categories of tasks are discussed below:

a. Airborne radioactivity associated with the cutting of the reactor coolant pipes is considered as an upper bound for all tasks to be performed during the repair effort. During the cutting, deposited activity in the vicinity of the cuts may become airborne. Conservatively assuming that all the activity in the vicinity of the cuts becomes airborne, workers inside the contamination control envelopes would be exposed to inhalation airborne radioactivity concentrations which are approximately 0.195 MPC, based on a 40 hr/wk (2000 hrs/yr) MPC for airborne concentrations listed in 10 CFR 20; Appendix B, Table 1, Column 1. The evaluation considered a protection factor associated with the workers wearing air line, full-face, pressure-demand respirators. For those tasks being performed inside containment during the cutting of the reactor coolant pipes it is conservatively estimated that the workers would be exposed to inhalation airborne radioactivity concentrations which are approximately 0.26 MPC based on 40 hr/wk (2000 hrs/yr) for airborne concentrations listed in 10 CFR 20. This value (0.26) is conservative because pipe cut operations do not occur continuously throughout the repair and workers will not be exposed to such airborne concentrations for 40 hrs/wk. Furthermore, all of the activity in the vicinity of the pipe cut is not likely to become airborne. 2

b. It is also expected that during the removal of contaminated nonmetallic insulation there may be localized airborne activity. However, FPL health physics procedures require workers removing such insulation to wear respirators as a precautionary measure;

therefore, the airborne radioactivity to which these workers will be exposed is expected to be insignificant. Furthermore, experience at the plant indicates that containment airborne radioactivity does not show a measurable increase, except locally, during insulation removal.

As discussed in Section 7.5, repair of the steam generators can be accomplished in one of three alternative ways, defined therein as Schemes I, II and III. In Schemes II and III, the steam generators may be removed and replaced in one piece. Thus, the man-rem doses associated with Schemes II and III would be similar to those given for Scheme I, except that the man-rem doses associated with the tasks related to the cutting of the steam generator at the transition cone do not apply. Since these two alternative schemes assume removal of the entire steam generator, the four major tasks associated with the cutting of the steam generators at the transition cone do not apply. These are:

a. Installation of steam generator support clips (for support of the steam dome assembly during the upper shell cut).

b. Preparation for steam generator upper shell cut.

c. Steam generator upper shell cut.

d. Steam generator upper internals cut and removal.

It is estimated that these tasks could result in a radiation exposure of approximately 250 man-rem for three steam generators; therefore, the range in man-rem dose associated with the removal/replacement effort for Schemes II and III is 400-1200 for three steam generators as compared to 650-1450 for Scheme I. 2

There are other tasks which vary between schemes, such as:

a. Scheme I - removal and replacement of concrete shield wall.

b. Scheme II - removal and replacement of a section of the containment wall.

c. Scheme III - removal and replacement of a section of the containment dome.

However, these tasks are of considerably less significance from a man-rem dose standpoint than those associated with the steam generator transition cone cut. Therefore, the man-rem range given above for Schemes II and III provides a reasonable estimate of total exposure.

3.3.7.3 Conclusions

In addition to meeting the quarterly, annual and cumulative occupational doses specified in 10 CFR 20.101 for any individual associated with the repair, the man-rem assessments presented in Subsections 3.3.7.1 and 3.3.7.2 were evaluated in order to determine whether or not the radiation received by personnel during the repair is ALARA. Although the man-rem associated with the repair effort are greater than the man-rem associated with the 1976 tube plugging, eddy current testing and other inspection-related activities at Turkey Point, it has been determined that approximately 14,000 man-rem may 1

be saved over the lifetime of the plants by implementing the repair on both units. Therefore, implementation of the repair effort is in accordance with Regulatory Guide 8.8 (ALARA) guidelines. Man-rem doses associated with the optimum disposal alternative are not appreciable (see Section 3.4.4). Thus, disposal considerations do not alter this conclusion. The primary basis of Regulatory Guide 8.8 is the conservative assumption that a non-threshold linear relationship exists between dose and biological effects, independent of the dose rate. On this basis, the projected long term man-rem savings gained from the repair effort will result in a linear reduction in any biological risk that is assumed to be proportional to the dose. 1

3.4 DISPOSITION OF STEAM GENERATOR LOWER ASSEMBLIES

The lower assemblies to be removed from Units 3 and 4 represent the single largest source of solid radioactive waste to be disposed of during the repair effort. The disposal effort is independent of the repair and is evaluated on that basis.

The primary side surfaces of the steam generators are contaminated by a tenacious film of deposited radioactive corrosion products made up primarily of iron, cobalt and manganese isotopes. Based on actual Turkey Point data provided in Section 5.2.2, it is estimated that at the time the lower assemblies are removed, each will contain approximately 250 curies of deposited gamma activity. 3

3.4.1 Objectives of Handling/Disposal Operations

The objectives of handling/disposal operations are as follows:

a. To dispose of the lower assemblies safely and economically

b. To provide means to handle/dispose of the steam generator lower assemblies so that radiation exposures to plant and contract personnel are as low as is reasonably achievable

c. To minimize the release of radioactivity to the environment so as to keep radiation exposure to the public as low as is reasonably achievable and within the limitations of 10 CFR 20

d. To package and ship the lower assemblies to a licensed land burial site in accordance with applicable state and federal regulations, including 10 CFR 71 and 49 CFR 170-178, or to store the lower assemblies onsite for decommissioning with the plant. 1

exposure to personnel since much of the expected radiation exposure is associated with the cutting up and packaging of the lower assemblies. If chemical decontamination is used, the liquid waste generated will be processed by one or more of the following: a) a mobile radwaste evaporator package, b) a mobile filter/demineralizer system, c) a mobile solidification unit. These mobile units are commercially available and have been used at several nuclear power plants. The solidification agent injection method involves filling the primary side of the steam generators with a suitable solidification agent prior to cutting. The solidification agent is used to fix any loose contamination in the primary side of the steam generators and to provide shielding to further reduce the radiation dose rates during the cutting operations.

The expected radiation exposure in man-rem associated with shipment preparation of the lower assemblies is presented in Section 3.4.4.

3.4.3.2 Preparation for Shipment by Barge

The steam generator lower assemblies will be sealed prior to removal from the containment so that the radioactivity will be contained within a strong, tight package, as required by 49 CFR 173. When the lower assemblies are to be shipped to a licensed land burial site, each one will be transported as a complete assembly to the barge facility and shipped as LSA material, in accordance with applicable state and federal regulations.

3.4.3.3 Shipment

Rail, truck, and barge transportation were investigated for shipment of the lower assemblies to a licensed land burial site.

Should rail transport of the lower assemblies be employed, each lower assembly will be cut up and packaged in approximately seven packages. The tubesheet and channel head of each lower assembly will make up one package, the tubes will make up approximately four packages, and the shell and other miscellaneous pieces will require another two packages. The heaviest load will be the channel and tubesheet combination which weighs approximately 80 tons. In addition, combination rail/truck shipments may be utilized; that is, rail would be used for the channel and tubesheet combination and truck would be used for the other pieces. If rail is utilized, rubber-tired tractors or tracked vehicles are required to transport the packages from the jobsite to a rail spur near Florida City, Florida (approximately 8 miles) and from the rail spur near the burial site at Barnwell, South Carolina (approximately 3 miles).

Should truck transport of the lower assemblies be employed, methods similar to those described above for rail transport will be used, except that the number of packages would have to be increased in order to meet the weight limitations for over-the-road shipments.

Should barge transport of the lower assemblies be employed, they may be shipped as complete assemblies. This method would involve the least amount of handling.

Regardless of the transportation method used, applicable requirements of 10 CFR 71 and 49 CFR 170-178 will be met. 1

3.4.4 Man-Rem Assessments

If the steam generator lower assemblies are shipped by rail and/or truck, they must be cut into suitably sized sections prior to shipment. Based on the radiation survey results given in Figure 3.3-7, it is estimated that the following man-rem doses per unit will be received during the preparation and shipment of a unit's lower assemblies for each of the given alternatives: 1

a.	Cutting up, packaging and shipping of the lower assemblies, assuming no decontamination and no solidification agent injection	750-1500 (man-rem)	2
b.	Cutting up, packaging and shipping of the lower assemblies, assuming solidification agent injection prior to cutting and a dose reduction factor of 2 is achieved	400-800 (man-rem)	2
c.	Cutting up, packaging and shipping of the lower assemblies, assuming decontamination of primary side surfaces prior to cutting and a DF of 10 is achieved	125-550 (man-rem)	2
d.	Cutting up, packaging and shipping of the lower assemblies, assuming 35 years storage prior to cutting	10-20 (man-rem)	1 2
e.	Onsite surveillance for 35 years and decommissioning of the steam generator lower assemblies	0.5-1.5 (man-rem)	2

The man-rem dose to workers by task for each of the above alternatives is contained in Table 3.4-3. A discussion of the uncertainties in predicting worker man-rem ranges for the alternatives is contained in Subsection 3.3.7.2. The man-rem dose to the public for each alternative is contained in Table 3.4-4. 2

If barge transport is employed, handling of the lower assemblies would be minimized and the resulting radiation exposure would be a small fraction of the man-rem doses associated with the alternatives given above.

It should be recognized that although the alternative which includes decontamination of the lower assemblies appears attractive from a man-rem savings standpoint, there is a significant cost penalty associated with it. Thus, the optimum cost/man-rem disposal alternative may not incorporate decontamination.

The man-rem dose associated with disposal of the steam generators is considered acceptable since a considerable man-rem savings is achieved by the repair effort (see Section 3.3.7.3).

3.4.5 Radioactive Releases and Dose Assessment Associated With Offsite Disposal |1

Radioactive airborne and liquid releases have been evaluated for the disposal effort using conservative, bounding parameters and assumptions. The total calculated release per unit for the disposal effort was found to be a small fraction of the 1976 radioactive releases per unit at Turkey Point and, therefore, is considered acceptable.

3.4.5.1 Airborne Releases

Radioactive airborne effluent releases to the environment resulting from the disposal effort have been estimated using the following assumptions and parameters:

a. Airborne releases are assumed to occur during the cutting operations, i.e., the cutting of the tubesheets, channel heads and tube bundles.

b. The primary side surfaces of the lower assemblies are expected to be contaminated primarily by deposited corrosion products. Typical corrosion product activities expected on the primary side surfaces after seven years of commercial operation are given in Table 5.2-1. (Seven years is based on the projected operation of Unit 3 prior to the repair effort. For Unit 4, the deposited corrosion product activity will be less since Unit 4 will have less operating time, a projection of 5 years.)

c. It is conservatively assumed that all the activity present in the vicinity of the cut will become airborne and be available for release to the environment.

d. Ninety days of radioisotope decay were assumed prior to cutting operations, based on the earliest reasonable time as dictated by the repair effort. No credit was taken for radioisotope 1
decay during cutting operations or for usage of water sprays.

e. The lower assemblies are assumed to be cut up in a facility which will be provided with HEPA filters in the exhaust line. The HEPA filters are conservatively assumed to be 99 percent efficient for capturing particulates.

Radioactive airborne effluent release to the environment based on the above assumptions is approximately 2.91 x 10^{-1} Ci for Unit 3. For Unit 4, this release is approximately 82 percent of the release for Unit 3. Details of the airborne effluent release by isotopes are given in Table 3.4-1. There are no radioactive airborne releases associated with onsite steam generator lower assembly storage, see Section 3.4.6. 1

3.4.5.2 Environmental Consequences of Airborne Releases

The critical organ and whole body doses for an adult at the worst site boundary location resulting from the estimated airborne effluent releases during the disposal effort were evaluated in the same manner as the repair effort described in Subsection 5.2.2.2. The critical organ (lung) and the whole body doses for an adult at the site boundary are estimated to be 3.13 x 10^{-2} mrem and 1.01 x 10^{-4} mrem, respectively during the disposal effort for Unit 3. The corresponding doses for Unit 4 would be 2.33 x 10^{-2} mrem and 7.82 x 10^{-5} mrem.

3.4.5.3 Comparison with Observed Gaseous Releases and Estimated Doses During Normal Operation

The estimated releases of radioactive airborne effluents per unit during the disposal effort are found to be much smaller than the observed gaseous effluent releases per unit for the Turkey Point Plant during 1976. Observed airborne effluent releases during 1976 are compared with estimated releases during the disposal effort in Table 3.4-2.

The critical organ (thyroid) and whole body doses to an adult at the worst site boundary location due to the release of gaseous effluents for the year 1976 were calculated to be 0.07 and 0.09 mrem/unit, respectively. The

estimated critical organ dose (lung) for the disposal effort is less than 46 percent of the calculated critical organ dose during 1976. The estimated whole body dose for the repair effort is less than 0.11 percent of the calculated whole body dose during 1976.

3.4.5.4 Liquid Effluent Releases

The only liquids associated with the disposal of the lower assemblies are those associated with decontamination. If the lower assembly primary side surfaces are decontaminated prior to cutting and the liquid wastes resulting from the decontamination operation are processed through a radwaste evaporator and subsequently discharged, the total radioactive liquid effluent release would be approximately 0.38 Ci/unit or about 9 percent of the observed total radioactive liquid release per unit during 1976. (Note that other methods of decontamination waste processing may be used which may slightly alter these results. These methods include solidification of the chemical solvents, filter/demineralization of the rinses, etc.)

3.4.6 Radioactive Releases and Dose Assessment Associated With Onsite Storage

The annual dose equivalent to any member of the public as a result of onsite storage of the used steam generator lower assemblies was evaluated. The evaluation considered a member of the public at the north site boundary for a period of 1 year.

As indicated in Section 3.4.2, prior to removal from the containment, the openings in the steam generator lower assemblies will be sealed to prevent the release of radioactivity during transfer and subsequent onsite storage. Since the lower assemblies will be completely sealed, there will be no airborne or liquid radioactive releases as a result of lower assembly onsite storage.

As discussed in Section 3.4.7, the radioactivity within the steam generators is immobile. Thus, if seal integrity were lost, releases to the environment are not likely. Nonetheless, a surveillance program will be implemented, comprised of a quarterly visual inspection of the external surfaces of the lower assemblies, area radiation surveys and random swipes of the welds sealing the covered openings in the lower assemblies. This surveillance program will provide further assurance that there are no unanticipated releases of radioactivity to the environment. 1

The only contribution, therefore, to the annual dose equivalent to any member of the public is from direct radiation emanating from the storage facility. The storage facility will be shielded, as required, in order to limit the dose rate at the outside limits of the storage facility to ≤ 2.5 mr/hr. The resulting dose equivalent to an individual at the north site boundary for a full year would be approximately 5.2×10^{-3} mrem, which is considered an insignificant contribution to the offsite dose. Furthermore, it is highly unlikely that

3.4.8 Conclusions

The steam generator lower assemblies will ultimately be disposed of in a licensed land burial site or decommissioned with the plant. Radiological considerations associated with each disposal alternative are acceptable. Thus, the final decision on the alternatives to be employed for storage, handling, 1
packaging and shipping of the lower assemblies will be based on economics.

3.4.9 References

1. "Decontamination of a PWR Primary System, SENA Plant," Volume 33, Proceedings of the American Power Conference, 1971.

2. "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH 1238, December 1972. 2

3.5 PLANT SECURITY

Specific plans for the physical protection of the Turkey Point Nuclear
Units during the steam generator repair will be addressed as required, in a 1
separate submittal withheld from public disclosure pursuant to paragraph 2.790(d), 10 CFR Part 2, Rules of Practice.

3.6 QUALITY ASSURANCE

The Quality Assurance Programs for FPL, Bechtel Power Corporation, and Westinghouse Electric Corporation are described in this section.

3.6.1 FPL Quality Assurance Program

The FPL Quality Assurance Program is described in "FPL Topical 3
Quality Assurance Report," (FPLTQAR 1-76A), Revision 2, September 8, 1977. The Topical Quality Assurance Report is an integral part of the corporate Quality Assurance Manual (FPL-NQA-100).

The following amplifications of FPLTQAR 1-76A are applicable for the Turkey Point Project.

FPL currently contemplates assuming responsibility for conducting the following on-site construction activities:

a. Field Procurement Functions

b. Field Quality Control Functions 1

c. Field Test Control Program

d. Field Special Processes

e. Field Quality Assurance Functions

3.6.2 Bechtel Power Corporation Quality Assurance Program

Home Office Engineering and miscellaneous services for the quality assurance program used by Bechtel for the Turkey Point Plant Project are in accordance with the applicable portions of "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, August 15, 1977. Responsibility for the Turkey Point Plant Project has been assigned to the Gaithersburg Power Division of the Bechtel Power Corporation.

The following amplifications of BQ-TOP-1 are applicable for the Turkey Point Project.

Bechtel is not responsible for conducting the activities specified in Section 3.6.1. 1

If the scope of work should change and Bechtel is assigned any or all of the functions of a. through e. of Section 3.6.1, these activities shall be performed in conformance with the applicable sections of Topical Report BQ-TOP-1, Revision 2A.

3.6.3 Westinghouse Electric Corporation Nuclear Energy Systems Division Quality Assurance Program

The quality assurance program used by Westinghouse during the design and fabrication of the steam generator lower assemblies is in accordance with "Westinghouse Nuclear Energy Systems Divisions Quality Assurance Plan," WCAP-8370. For activities which occurred during the period from January 1, 1975 to September 30, 1977, the program is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the program is presented 3
in WCAP-8370, Revision 8A.

3.7 REGULATORY GUIDE APPLICABILITY TO REPAIR PROGRAM

Section 2.1.4 discusses Regulatory Guide compliance during manufacture of the lower assemblies. Regulatory Guide applicability to repair program activities other than manufacture and other than those addressed in the FPL and Bechtel QA programs are discussed below.

1.31 Control of Stainless Steel Welding (Rev. 1, June 1973):

Control of stainless steel welding complies with the Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1, dated 11/24/75) except as discussed below:

1. Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2 and 3 components are controlled to deposit from 8 to 25 percent delta ferrite, except for 309 and 309L welding filler materials which are controlled to deposit from 5 to 15 percent delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. Use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required.

   These limits for delta ferrite in austenitic stainless steel welding materials comply with Interim Regulatory Guide 1.31 since the upper limit of 20 percent delta ferrite does not apply for welds that are not heat treated after welding (Paragraph 3b), except for solution heat treatment. Solution heat treatment, although not required after welding, is permitted in order to avoid sensitization.

   Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1977 edition, Paragraph NB-2433, except that an undiluted weld deposit is required for each heat of bare wire used with the Gas Metal-Arc (GMA) process.

2. Reference: Paragraph B.2 of the Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1977 edition.

3. Reference: Paragraph B.3.a of the Interim Position. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25 percent delta ferrite based on chemistry, except for 309 and 309L welding materials which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.

4. Reference: Paragraph B.3.b of the Interim Position. This paragraph is complied with for welding material certification.

5. Reference: Paragraphs B.4.a, .b, and .c of the Interim Position. Measurement of production welds for delta ferrite is not performed.

1.44 Control of the Use of Sensitized Stainless Steel (May 1973)

Subject to the following statements, the use of unstabilized austenitic stainless steel for components which are part of the reactor coolant pressure boundary, which are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions, complies with Regulatory Guide 1.44.

1. Reference: Paragraph C.1 of the Regulatory Guide. Contamination of austenitic stainless steels (Type 300 series) by compounds which could cause stress corrosion cracking is avoided during all stages for repair welding. Except for trichlorotrifluoroethan (TCTFE) which meets the requirements of Military Specification MIL-C-8130 2B, cleaning is limited to solutions which contain not more than 200 ppm of chlorides. Rinsing or flushing is with water containing less than 200 ppm of chlorides. Special rinsing techniques are used to assure complete removal of TCTFE when crevices or undrainable areas are present. Foreign substances in contact with austenitic stainless steel (dye, lubricants, penetrant materials, marking materials, masking tape, etc.) are controlled so as not to contain more than 200 ppm of chlorides, or are removed immediately following the operation in which they are used. Crevices and undrainable areas are protected prior to use of materials containing more than 200 ppm of chlorides. All substances in contact with austenitic stainless steel are removed prior to any elevated temperature treatment. 3

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with water which contains less than 200 ppm of chlorides. The influent water quality during final flushing or pre-operational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants.

1.71 Welder Qualification for Areas of Limited Accessibility (December 1973).

The response to this Regulatory Guide is as follows:

1. Reference: Paragraph C.1 of the Regulatory Guide. Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Welding conducted in areas of limited access is subjected to the required nondestructive testing and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted.

2. Reference: Paragraph C.2 of the Regulatory Guide. Requalification is required when any of the essential variables of ASME Section IX are changed, or when any authorized inspector questions the ability of the welder to perform satisfactorily the requirements of ASME Sections III or IX.

3. Reference: Paragraph C.3 of the Regulatory Guide. Production welding is monitored and welding qualifications are certified in accordance with 1. and 2. above.

1.123 Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (October 1976).

The FPL position regarding ANSI N45.2.13 is as stated in "FPL Topical Quality Assurance Report," FPLTQAR 1-76A, Revision 2, September 8, 1977, Appendix C. 3

The Bechtel position is as stated in "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report, BQ-TOP-1, Revision 2A, August 15, 1977, Introduction.

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 3 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description
1	3J01-3P200A/1	No	Power Cable to Reactor Coolant Pump A Motor
2	3J01-3P200A/2	No	Power Cable to Reactor Coolant Pump A Motor
3	3P200A-T3P52/1	No	Power Cable to RCP-A Motor Space Heater
4	3P232A-T3P52/1	No	Power Cable to RCP-A Lift Oil Pump Motor
5	3V30A-T3P41/1	Yes	Power Cable Emergency Containment Cooling Fan 3A
6	CV853A-TB3141/1	No	Control Cable to Accumulator Vent Valve 3V-853A 7
7	FIC156-T3C21/1	No	Reactor Coolant Pump A, #1 Seal B/P Flow
8	FIC490-T3C21/1	No	RTD Bypass Flow
9	LC3417-T3C21/1	No	Reactor Coolant Pump A Upper Level Oil Reservoir
10	LC406A-T3C21/2	No	Reactor Coolant Pump A Stand Pipe Level
11	LC406B-T3C21/2	No	Reactor Coolant Pump A Stand Pipe Level
12	LS1570-T3C22/1	No	NPSH for North Containment Sump #3, Recirculation
13	MOV535-T3C22/1	No	PREZ-Relief Isolation Valve Control Cable
14	MOV535-T3P51/1	No	PREZ-Relief Isolation Valve Power Cable
15	PC417-T3C21/1	No	Reactor Coolant Pump A Oil Lift Pump Control Cable
16	PI131B-PT131/1	No	Reactor Coolant Pump A Shaft Seal Differential Pressure
17	PI156B-PT156/1	No	Reactor Coolant Pump A Seal Injection Indicator
18	RI1403-T3P52/1	No	Area Radiation Monitor Indicator
19	T3122-FT156A/1	No	Reactor Coolant Pump-A Seal Injection High Range

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 3 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description
20	T3122-FT156B/1	No	Reactor Coolant Pump-A Seal Injection Low Range
21	T3122-LT477/1	No	Steam Generator A Wide Range Level
22	T3122-LT920/1	No	Accumulator Tank A Level
23	T3122-PT921/1	No	Accumulator Tank A Pressure
24	T3123-FT416/1	Yes	Reactor Coolant Flow - Loop A
25	T3123-LT476/1	Yes	Steam Generator A Channel III Narrow Range Level
26	T3124-FT415/1	Yes	Reactor Coolant Flow - Loop A
27	T3124-LT475/1	Yes	Steam Generator A Channel II Narrow Range Level 7
28	TB3116-LT920/1	No	Accumulator Tank A Level
29	TB3118-FIC156/1	No	Reactor Coolant Pump A Seal Injection
30	TB3118-FT156A/1	No	Reactor Coolant Pump A Seal Injection High Range
31	TB3118-FT156B/1	No	Reactor Coolant Pump A Seal Injection Low Range
32	TB3125-T3C13/1	Yes	Charcoal Filter Spray Valve for Emergency Containment Cooling Filter 3A
33	TB3127-T3C22/1	Yes	Charcoal Filter Spray Backup Valve Emergency Containment Cooling Filter 3A
34	TB3141-T3C11/2	Yes	PREZ-Relief Solenoid Valve 3V-455C Control Cable
35	TB3141-T3C11/3	Yes	Excess Letdown Transfer Valve Drain Tank/VCT
36	TB3141-T3C11/4	Yes	Exchanger Let Down Valve
37	TB3141-T3C22/1	No	Reactor Coolant Drain Tank Pump 3A Suction Valve 3V1003A

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 3 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description
38	TB3141-T3C22/2	No	Reactor Coolant Drain Tank Pump 3A Suction Valve 3V1003A
39	TB3303-LT474/1	Yes	Steam Generator A Channel I Narrow Range Level
40	3V2A-T3P11/1	No	Reactor CRDM Cooler 3A Power Cable
41	3V2A-T3P53/1	No	CRDM Mechanical Cooler Motor Heater 3A Power
42	3V2B-T3P12/1	No	Reactor CRDM Cooler 3B Power Cable
43	3V2B-T3P53/1	No	CRDM Mechanical Cooler Motor Heater 3B Power

7

TABLE 3.3-2

ESTIMATED MAN-REM DOSE TO WORKERS FOR THE REPAIR OF THREE STEAM GENERATORS

	Task Description	Average Radiation Field (rem/hr)	Estimated Man Hours in Radiation Fields (hrs)	Task Man-Rem Dose (man-rem)
1.	Erection of scaffolding and pipe supports at the steam generators and piping to be cut	0.005-0.1	1620	92.1
2.	Removal of insulation from RCS piping and steam generator	0.005-0.1	1500	63
3.	SG and RCS pipe surface decontamination	0.005-0.1	1200	61.5
4.	Installation of steam generator support clips	0.005-0.1	1200	40.5
5.	Preparation for steam generator upper shell cut	0.005-0.1	1350	48
6.	Steam generator upper shell cut	0.005-0.1	2100	45
7.	Steam generator upper internals cut and removal	0.005-0.3	1440	102.6
8.	Preparation of RCS hot legs for cut	0.05-0.1	300	22.5
9.	RCS hot leg cut	0.005-0.1	780	29.1
10.	Preparation of RCS pump legs for cut	0.05-0.1	300	22.5
11.	RCS pump leg cut	0.005-0.1	780	29.1
12.	Installation of steam generator cover plates	0.005-0.1	780	37.5
13.	Removal of steam generator lower assemblies to storage facility	0.005-0.1	1140	39
14.	Installation of new lower assemblies, concrete removal and replacement, laydown facilities erection, temporary relocation of containment equipment, other pipe cuts and welds, etc.	0.005-0.05	34,000	260
15.	Special crews and indirect work (e.g., scaffolding, cleanup, temporary power, etc.)	0.005-0.05	23,800	209
16.	Miscellaneous(1) (e.g., supervisory field engineering, security, QA/QC and health physics activities)	0.005-0.05	22,000	200
	Total for repair of three steam generator lower assemblies			1301.4
	Estimated range for repair of three steam generator lower assemblies			650-1450

2

NOTE:

(1) Miscellaneous covers tasks which may be related to the removal as well as to the installation of the steam generators.

TABLE 3.4-1

ESTIMATES OF AIRBORNE RELEASES
TO ENVIRONMENT DURING STEAM GENERATOR
LOWER ASSEMBLY DISPOSAL EFFORT (1)(2)

Isotope	Release (Ci)
Co-58	7.95×10^{-2}
Co-60	1.3×10^{-1}
Mn-54	1.23×10^{-2}
Fe-59	8.65×10^{-4}
Cr -51	1×10^{-3}
Zr-95	6.44×10^{-3}
Nb-95m	1.37×10^{-4}
Nb-95	1.18×10^{-2}
Ru-103	2.34×10^{-3}
I-131	1.44×10^{-4}
Ba-137m	4.69×10^{-4}
Cs-137	4.99×10^{-4}
Ce-141	8.22×10^{-4}
Ce-144	2.33×10^{-2}
Pr-144	2.23×10^{-2}
Total	2.91×10^{-1}

NOTES (1) Disposal effort releases are given for Unit 3. For Unit 4 the releases will be less, since Unit 4 will have less operating time and, therefore, less original contamination.

(2) Releases less than 10^{-4} are of no significance and, therefore, are not listed.

TABLE 3.4-2

COMPARISON OF GASEOUS EFFLUENT RELEASES

Isotope	Average 1976 Release/Unit (Ci)	Estimated Release/Unit During SG Disposal Effort (Ci) (2)
Noble gases	7800	Negligible
Iodines	0.26	1.44×10^{-4}
Particulates (1)	0.038	2.91×10^{-1}
Tritium	2.6	Negligible

Notes

(1) Approximately 55.2 percent and 12.3 percent of the particulate total release during the year 1976 are Co-58 and Co-60, respectively.

(2) Disposal effort releases are given for Unit 3. For Unit 4 the releases will be less, since Unit 4 will have less operating time and, therefore, less original contamination.

TABLE 3.4-3

ESTIMATED MAN-REM DOSE TO WORKERS FOR STEAM GENERATOR DISPOSITION ALTERNATIVES

Task	Man-rem estimate for cutting up, packaging, and shipping of the steam generator lower assemblies, assuming no decontamination and no solidification agent injection	Man-rem estimate for cutting up, packaging, and shipping of the steam generator lower assemblies, assuming solidification agent injection prior to cutting and a dose reduction factor of 2 is achieved	Man-rem estimate for cutting up, packaging, and shipping of the steam generator lower assemblies, assuming decontamination of primary side surfaces prior to cutting and a DF of 10 is achieved	Man-rem estimate for cutting up, packaging, and shipping of the steam generator lower assemblies, assuming 35 years storage prior to cutting	Man-rem estimate for onsite surveillance for 35 years and decommissioning of the steam generator lower assemblies
1. Preparation for cutup at the storage facility	Negligible	Negligible	Negligible	Negligible	NA
2. Placement of temporary shielding at the lower assemblies	220	NA	NA	NA	NA
3. Solidification agent injection	NA	20	NA	NA	NA
4. Allowance for cutup problems due to solidification agent injection	NA	60	NA	NA	NA
5. Decontamination operations, including processing of decontamination waste	NA	NA	150	NA	NA
6. Surveillance for 35 years	NA	NA	NA	1.1	1.1
7. Cutup and handling	1160	615	100	12.2	NA
8. Packaging and shipping	105.7	52	12	1.1	NA
9. Decommissioning	NA	NA	NA	NA	0.2
10. Miscellaneous support activities by manual crafts at the steam generators (e.g., clean up, erect and remove scaffolding, temporary lighting, etc.)	0.5	0.15	0.04	0.04	0.02
11. Supervisory and nonmanual time at the steam generators (e.g., health physics, security, field engineering, etc.)	14	6	1.2	0.13	0.04
Total for the Alternative	1500.2	735.15	285.24	14.57	1.36
Estimated range for the Alternative	750 - 1500	400 - 800	125 - 550	10 - 20	0.5 - 1.5

2

TABLE 3.4-4

ESTIMATED MAN-REM DOSE TO THE PUBLIC FOR STEAM GENERATOR DISPOSITION ALTERNATIVES

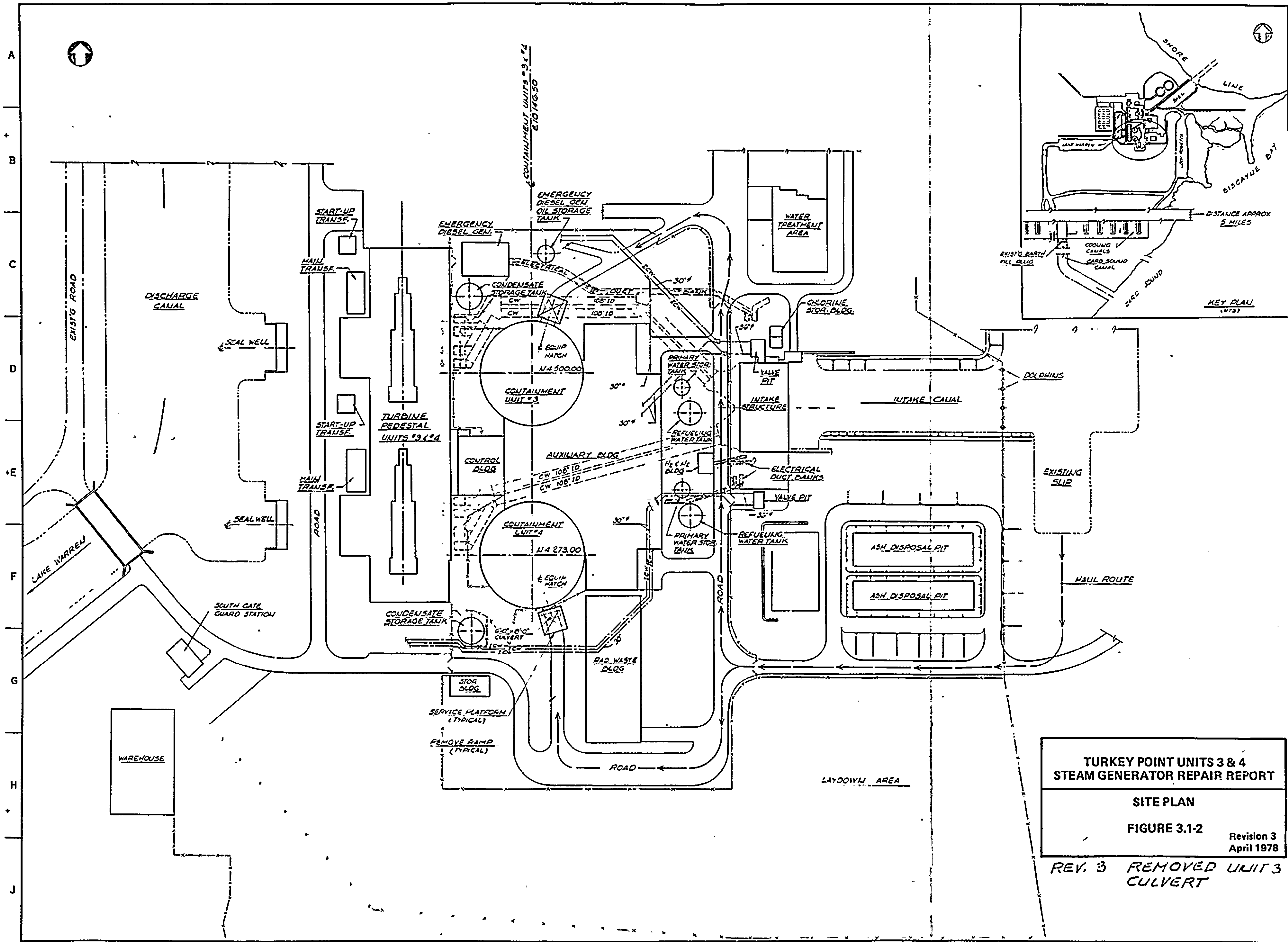
Steam Generator Disposition Alternative		Cumulative Whole Body Dose at the Site Boundary (1) (man-rem)	Critical Organ (Lung) Dose at the Site Boundary (1) (man-rem)	Cumulative Direct Dose During Transport (2) (man-rem)
a.	Cutting up, packaging and shipping of six steam generator lower assemblies, assuming no decontamination and no solidification agent injection.	2.8×10^{-6}	5.5×10^{-5}	2.1
b.	Cutting up, packaging and shipping of six steam generator lower assemblies, assuming solidification agent injection prior to cutting.	1.5×10^{-6}	5.5×10^{-5}	1.2
c.	Cutting up, packaging and shipping of six steam generator lower assemblies, assuming decontamination of the steam generators.	2.8×10^{-7}	5.5×10^{-6}	1.2
d.	Cutting up, packaging and shipping of six steam generator lower assemblies, assuming 35 years of storage prior to cutting.	3.7×10^{-5}	4.1×10^{-7}	0.9
e.	Store steam generators on site for 35 years and decommission with plant.	3.7×10^{-5}	NONE	NA

2

NOTES:

(1) The site boundary (4164 feet) was chosen for the cumulative direct and particulate doses associated with storage, cutup and packaging of the steam generator lower assemblies for conservatism. It is obvious that no individual will be at the site boundary for the duration. The closest permanent resident to the site is approximately five miles away.

(2) Public doses during transportation of the steam generator pieces are based on the conservative guidelines provided in Reference 2.



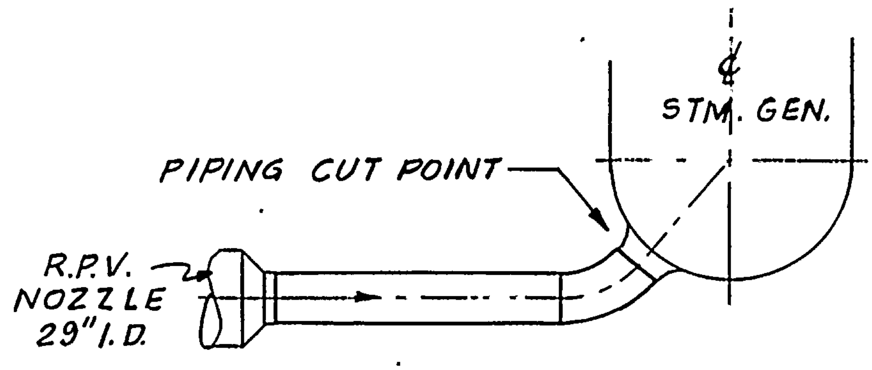
TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

SITE PLAN

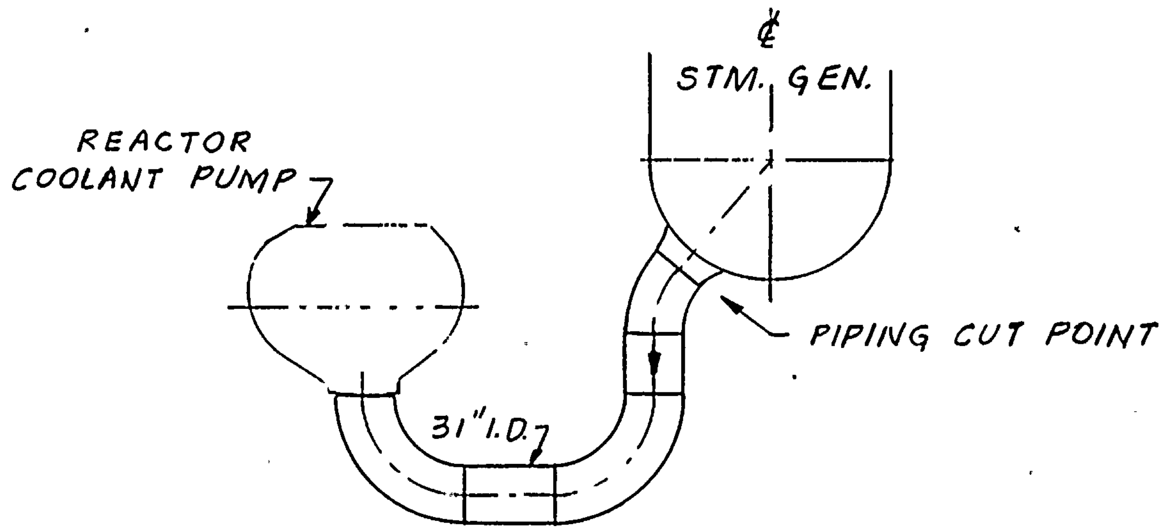
FIGURE 3.1-2

Revision 3
April 1978

REV. 3 REMOVED UNIT 3 CULVERT



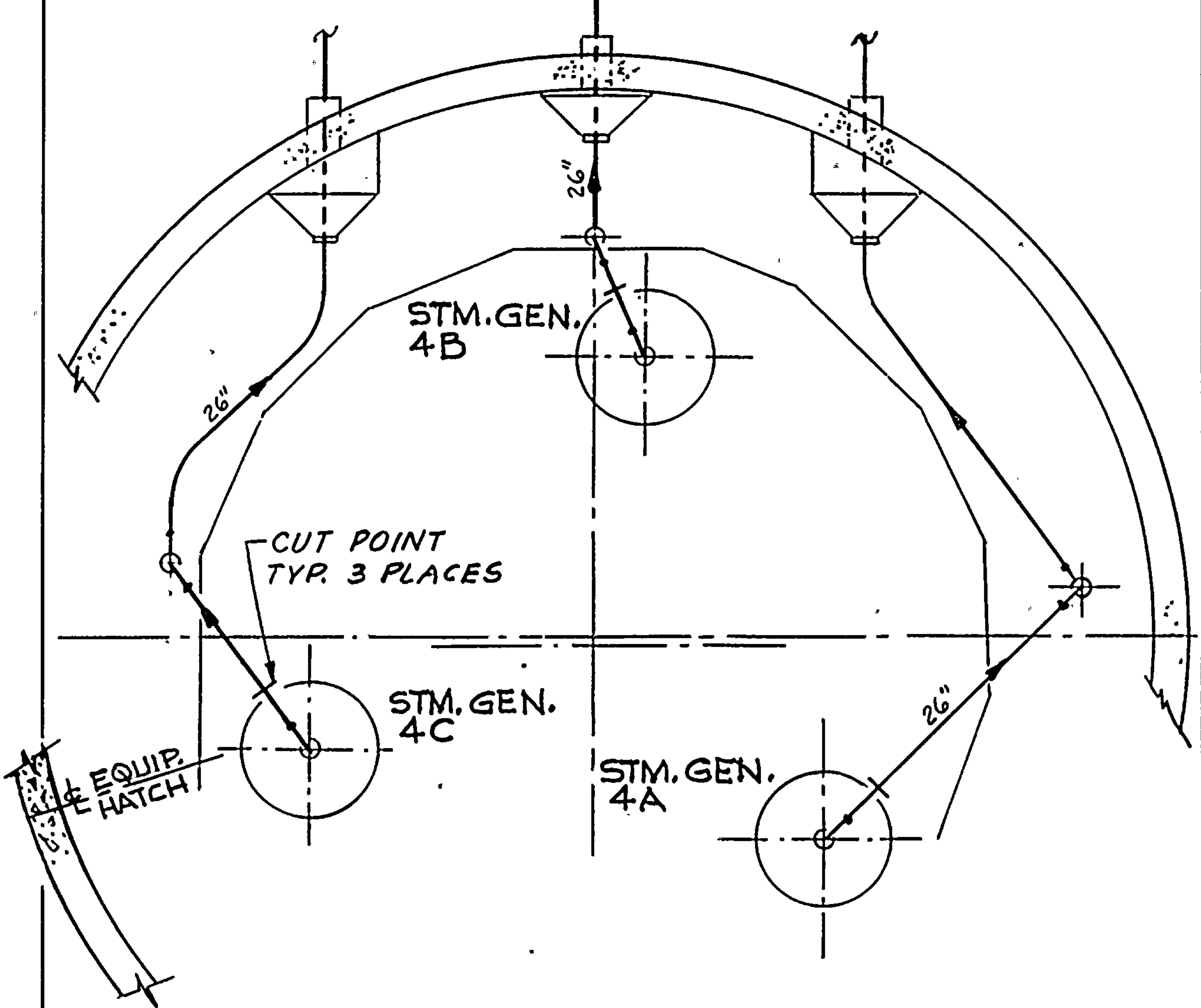
SECTION
REACTOR COOLANT PIPING
HOT LEG



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

REACTOR COOLANT PIPING
CUT POINTS

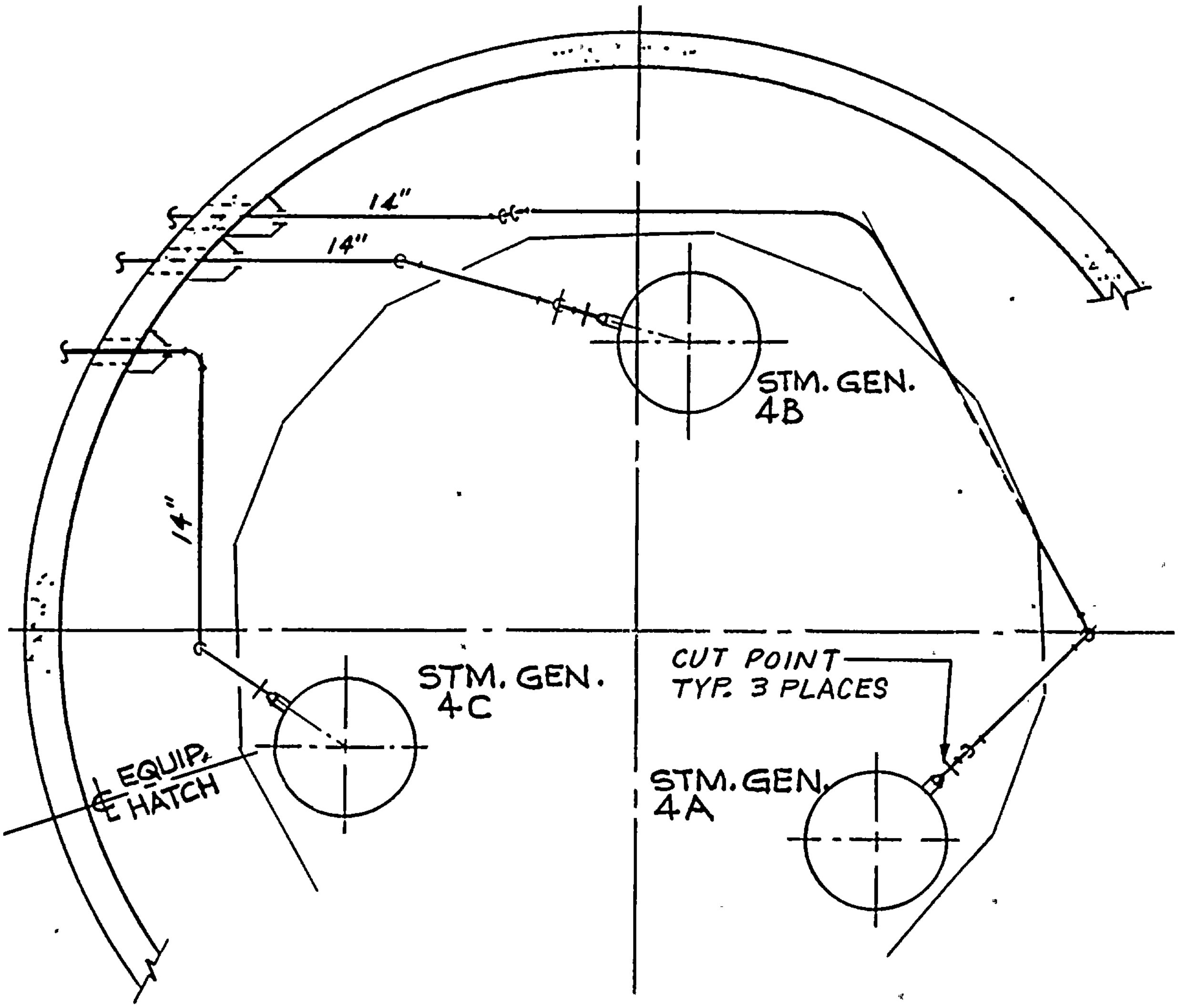
FIGURE 3.2-1



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MAIN STEAM PIPING CUT POINTS

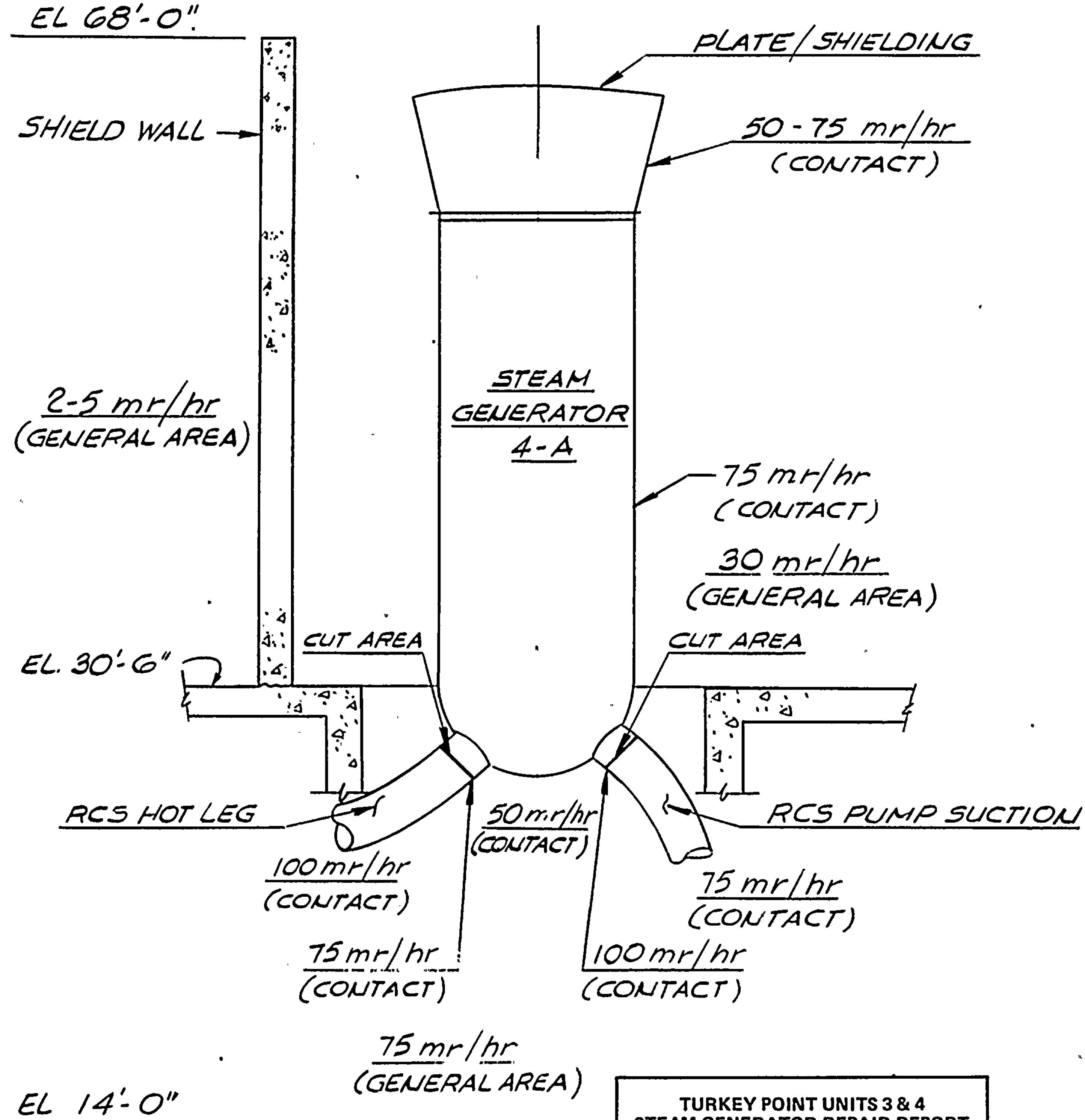
FIGURE 3.2-2



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MAIN FEEDWATER PIPING CUT POINTS

FIGURE 3.2-3



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

DOSE RATES AROUND STEAM GENERATOR 4A

FIGURE 3.3-7

4.0 RETURN-TO-SERVICE TESTING

Following steam generator repair, a preoperational testing program will be conducted as required to provide the necessary assurance that the facility can be operated in accordance with design requirements and in a manner that will not endanger the health and safety of the public.

rigging incidents, an evaluation was performed conservatively assuming that loss of function would occur. Locations for postulated transportation incidents are provided below. Figure 5.2-1 illustrates these locations, as well as the haul route from the existing barge slip. The accident locations evaluated are based on haul routes associated with each onsite receipt and disposition alternative being considered for the lower assemblies. It must be noted, however, that these evaluations are provided to illustrate capability only; actual construction practices will preclude occurrence of these accidents. 3

LOCATION B - Unit 4 buried intake cooling water (ICW) piping under Unit 4 equipment hatch construction area

LOCATION C - Unit 4 buried ICW piping and duct bank under steam generator haul route

LOCATION D - Unit 4 buried duct bank under steam generator haul route

LOCATION E - Unit 3 buried ICW piping under steam generator haul route

LOCATION F - Unit 3 buried duct bank under steam generator haul route

LOCATION G - Unit 3 buried ICW piping under Unit 3 equipment hatch access road

LOCATION H - Unit 3 buried duct bank under Unit 3 equipment hatch construction area

Other plant equipment adjacent to the transportation haul routes was evaluated and determined to be either not required for safe shutdown or far enough from the haul route to preclude the potential for damage.

5.2.1.4 Evaluation of Postulated Transportation Incidents

The safety analysis for each location provided in this Subsection includes the analysis of the assumed loss of all safety-related functions at each location.

<u>Unit 4 Shutdown/Under Repair - Unit 3 Operating</u>

LOCATION B - Unit 4 Buried ICW Piping Under Unit 4 Equipment Hatch Construction Area

It was conservatively assumed that all Unit 4 ICW piping in this location was damaged, such that total cooling function to the Unit 4 component cooling water (CCW) heat exchangers was lost. Since, under these conditions, Unit 4 is shutdown, the only potential safety-related consequence is loss of cooling to the Unit 4 spent fuel pool.

As discussed above in the scenario with only Unit 3 operating, Unit 3 would be maintained in the hot shutdown condition following reactor trip with secondary heat removal via the auxiliary feedwater system. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling to the Unit 3 spent fuel pool and other equipment as required. The Unit 4 component cooling water system has sufficient capacity to supply its own operational cooling needs and to supply the hot shutdown requirements of Unit 3.

LOCATION F - Unit 3 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all power and control cables to the Unit 3 ICW pumps were damaged such that total cooling function to the Unit 3 CCW heat exchangers was lost. 5

Unit 3 would be tripped and maintained in a hot shutdown condition. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling as required.

Loss of function of the remaining cables in this duct could require that Unit 4 also be brought to hot shutdown condition until an alternate power feed is reestablished to the Unit 4 circulating water pumps. Unit 4 ICW system would remain functional to supply the Unit 3 and Unit 4 hot shutdown and spent fuel pool cooling requirements. Following this, Unit 4 can be returned to full power operation.

5.2.1.5 References

1. "Design of Structures for Missile Impact," BC-TOP-9A, Revision 2, September 1974 1

5.2.2 Radioactive Releases and Dose Assessment

Radioactive airborne and liquid releases have been evaluated for the repair effort using conservative, bounding parameters and assumptions. In order to assess the significance of these releases, they were compared with the radioactive releases at Turkey Point in the year 1976. The total calculated release per unit for the repair effort was found to be a small fraction of the 1976 radioactive releases per unit and, therefore, is considered acceptable.

5.2.2.1 Airborne Releases

Airborne effluent releases to the environment resulting from the repair effort have been estimated using the following assumptions and parameters.

a. Airborne releases are assumed to occur during the cutting operations, i.e., the cutting of the reactor coolant pipes. Cutting and removal of concrete necessary for this operation are expected to result in insignificant releases of radioactivity.

b. The repair effort is assumed to start after approximately 5 and 7 years of commercial operation for Unit 4 and 3, respectively.

c. The steam generators and the reactor coolant pipes are expected to be contaminated primarily by deposited corrosion products. Typical corrosion product activities expected on the steam generator primary side surfaces after 7 years of commercial operation are given in Table 5.2-1 (7 years is based on the projected operation of Unit 3 prior to the repair effort. For Unit 4, the deposited corrosion product activity will be less, since Unit 4 will have less operating time, a projection of 5 years.)

d. It is conservatively assumed that all the activity present in the vicinity of the cut will become airborne and be available for release to the environment.

e. Fifteen days of radioisotope decay were assumed prior to cutting the reactor coolant pipes.

f. During cutting, the reactor coolant pipes will be surrounded by radioactivity control envelopes whose atmosphere will be exhausted through HEPA filters. The HEPA filters are conservatively assumed to be 99 percent efficient for capturing the particulates.

Radioactive airborne effluent release to the environment based on the above assumptions is approximately 1.06×10^{-2} Ci for Unit 3. For Unit 4, this release is approximately 89 percent of the release for Unit 3. Details of the airborne effluent release by isotopes are given in Table 5.2-2.

5.2.2.2 Environmental Consequences of Airborne Releases

The critical organ and whole body doses for an adult at the worst site boundary location resulting from the estimated airborne effluent releases during the repair effort were evaluated using a ground level release atmospheric dispersion factor of 1.02×10^{-6} sec/m^3 (Turkey Point FSAR Chapters 2 and 11) and the dose models and dose factors given in Regulatory Guide 1.109. The critical organ (lung) and the whole body doses for an adult at the site boundary are estimated to be 7.12×10^{-4} mrem and 2.78×10^{-6} mrem respectively during the repair effort for Unit 3. The corresponding doses for Unit 4 would be 5.45×10^{-4} mrem and 2.32×10^{-6} mrem.

5.2.2.3 Comparison with Observed Gaseous Releases and Estimated Doses During Normal Operation

The estimated releases of radioactive airborne effluents per unit during the repair effort are found to be much smaller than the observed gaseous effluent releases per unit for the Turkey Point Plant during the year 1976. Observed gaseous effluent releases during 1976 are compared with estimated releases during the repair effort in Table 5.2-3.

The critical organ (thyroid) and whole body doses to an adult at the worst site boundary location due to the release of gaseous effluents for the year 1976 were calculated to be 0.07 and 0.09 mrem/unit, respectively. The estimated critical organ (lung) dose for the repair effort is less than 1.1 percent of the calculated critical organ dose during 1976. The estimated whole body dose for the repair effort is less than 0.003 percent of calculated whole body dose during 1976.

5.2.2.4 Liquid Effluent Releases

Liquid effluent releases resulting from the repair effort were estimated using the following parameters and assumptions:

a. The reactor coolant system is drained 15 days after reactor shutdown and the reactor coolant is subsequently discharged after

processing through a mixed bed demineralizer and through the boric acid recovery evaporator as required. Laundry waste water is discharged without processing.

b. The decontamination factors for processing equipment are listed below and are in accordance with NRC NUREG-0017:

Processing Equipment	Decontamination Factors		
	Iodines	Cs&Rb	Others
Mixed bed demineralizer	10	2	10
Boric acid recovery evaporator	100	1000	1000

c. Reactor coolant concentrations are given in Table 5.2-4 and are based on radiochemical analyses taken during 1976 at Turkey Point.

d. The mass of the reactor coolant discharged after processing is 4.08×10^5 lbs.

e. Laundry releases were estimated using the expected specific activities in the laundry waste water given in Table 5.2-5 and assuming approximately 22,000 gal/day of laundry waste water will be discharged for approximately 300 days during the repair effort for one unit. (It is expected, however, that on the average only 10,000 gal/day of laundry waste water will be discharged during this period.)

The total radioactive liquid effluent release based on the above assumptions is estimated to be approximately 0.55 Ci/unit, excluding tritium and dissolved gases, and approximately 185 Ci/unit of tritium. Details of this release by isotope are given in Table 5.2-6.

5.2.2.5 Comparison with Observed Radioactive Liquid Releases During Normal Operation

Estimated radioactive liquid releases during the repair effort are compared with the observed liquid waste releases during the year 1976 in Table 5.2-7. The estimated total radioactive liquid release per unit (excluding tritium and dissolved gases) during the repair effort is seen to be about 13 percent of the observed total liquid waste release per unit (excluding tritium and dissolved gases) during 1976. The estimated tritium release per unit during the replacement effort is about one half the observed tritium release per unit during 1976.

TABLE 5.2-1

ESTIMATED CORROSION PRODUCT ACTIVITIES ON STEAM GENERATOR PRIMARY SIDE SURFACES (1) (2) (3)

Isotope	Activity ($\mu Ci/cm^2$)	Isotope	Activity ($\mu Ci/cm^2$)
Co-58	26.03	I-131	4.5×10^{-1}
Co-60	18.1	I-132	4.5×10^{-1}
Mn-54	2.03	Te-132	4.5×10^{-1}
Fe-59	4.65×10^{-1}	Cs-137	6.74×10^{-2}
Cr-51	1.35	Ba-140	4.5×10^{-1}
Zr-95	2.25	La-140	1.35
Nb-95	3.15	Ce-141	7.5×10^{-1}
Mo-99	4.5×10^{-1}	Ce-144	3.72
Tc-99m	2.25×10^{-1}	Np-239	4.8
Ru-103	1.5		

Notes

(1) The activities are based on actual Turkey Point data

(2) Activities listed are extrapolated to 7 years of commercial operation

(3) For Unit 4 (approximately 5 years of commercial operation) activities of long lived isotopes Co-60, Cs-137, Mn-54 and Ce-144 are expected to be approximately 0.72 of their corresponding activities for Unit 3 (approximately 7 years of commercial operation)

TABLE 5.2-2

ESTIMATED AIRBORNE RELEASES TO ENVIRONMENT DURING THE STEAM GENERATOR REPAIR (1) (2)

Isotope	Release (Ci)
Co-58	3.28×10^{-3}
Co-60	2.63×10^{-3}
Mn-54	2.87×10^{-4}
Cr-51	1.34×10^{-4}
Zr-95	2.82×10^{-4}
Nb-95	4.2×10^{-4}
Ru-103	1.69×10^{-4}
I-131	1.81×10^{-3}
I-132	2.77×10^{-4}
Ce-144	5.25×10^{-4}
Pr-144	5.25×10^{-4}
Total	1.06×10^{-2}

Notes

(1) Repair effort releases are given for Unit 3. For Unit 4 the releases will be less, since Unit 4 will have less operating time and, therefore, less original contamination.

(2) Releases less than 10^{-4} are of no significance and, therefore, are not listed.

TABLE 5.2-3

COMPARISON OF GASEOUS EFFLUENT RELEASES

Isotope	Average 1976 Release/Unit (Ci)	Estimated Release/Unit During the SG Repair Effort (Ci) (2)
Noble gases	7800	Negligible
Iodines	0.26	2.09×10^{-3}
Particulates (1)	0.038	8.5×10^{-3}
Tritium	2.6	Negligible

Notes

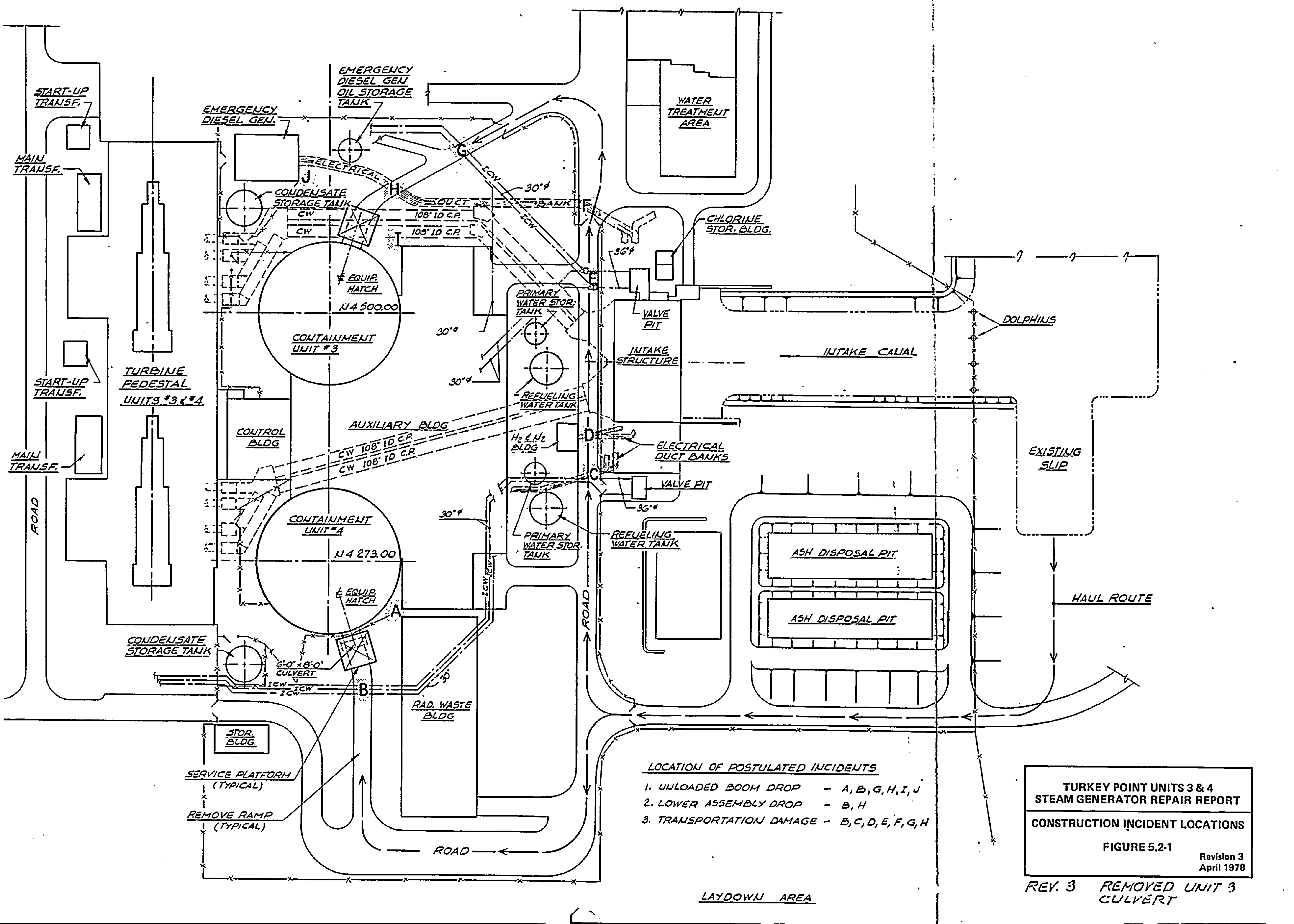
(1) Approximately 55.2 percent and 12.3 percent of the particulate total release during the year 1976 are Co-58 and Co-60, respectively.

(2) Repair effort releases are given for Unit 3. For Unit 4 the releases will be less, since Unit 4 will have less operating time and, therefore, less original contamination.

TABLE 5.2-7

COMPARISON OF RADIOACTIVE LIQUID EFFLUENT RELEASES

Isotope	Average 1976 Release/Unit (Ci)	Estimated Release During the S.G. Repair Effort/Unit (Ci)
Total (excluding tritium and dissolved gases)	4.3	0.55
Tritium	386	185



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CONSTRUCTION INCIDENT LOCATIONS

FIGURE 5.2-1

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REV. 3 REMOVED UNIT 3 CULVERT

6.3.2 Laundering Operations

Laundry waste water generated during the repair activities will originate from facilities built for the proposed effort. If required, laundry waste water will be directed to the liquid radwaste system for processing (see Subsection 3.3.6.3). Additional information on the expected quality and quantity of laundry waste water from the steam generator repair program is provided in Subsection 5.2.2.4 of this report.

6.4 CONSTRUCTION

Construction activities at the time of the repair effort will satisfy applicable laws that are in force at that time. These activities will have a negligible effect on noise levels, dust or smoke.

6.4.1 Noise

To examine order of magnitude sound pressure levels (SPL), typical construction noise sources were assumed and their SPL at the site boundary (0.8 miles) was calculated. The calculation utilized the methods described in References 1 and 2. Attenuation results are summarized in Table 6.2-2. Based on the results of Table 6.2-2 and based on the location of the site in a low population area and the limited amount of construction equipment required, noise resulting from the repair program for the steam generators is expected to have negligible, additional impact on the local area.

To protect personnel located on the site, Occupational Safety and Health Administration Standards (OSHA) will be followed. 1

6.4.2 Dust

Dust created by movement of vehicular traffic in an unpaved area, if any, will be abated by periodically spraying with water. The frequency of spraying and the quantity of water sprayed will be determined by visual inspection of the areas and will vary with the weather conditions.

6.4.3 Open Burning

Open burning is not anticipated during the steam generator repair effort. However, should the necessity arise, applicable county and state regulations for open burning will be followed.

6.5 RADIOLOGICAL MONITORING

The estimated releases of radioactive airborne and liquid effluents during the repair effort are found to be much smaller than observed effluent releases for the operating plant during 1976. The comparison is shown in Section 5.2.2. The radioactive effluent release points during steam generator repair activities will be the same as during normal plant operations; therefore, the plant radioactive process monitor locations will not be affected.

Since releases of radioactive effluents during the repair program will be small relative to the operating plant and their potential exposure pathway will be the same as for the existing plant operation, these effluents will be monitored in accordance with FPL's existing environmental monitoring program.

6.6 RETURN TO OPERATION

6.6.1 Water Use

Water consumption during post repair plant operation is expected to be considerably less than water consumption during current plant operation. Currently, frequent shutdowns of the nuclear units to perform steam generator tube plugging and/or eddy current inspection result in a significant water consumption. Steam generator filling and draining operations are required to locate the leaky tubes prior to plugging, for hydrostatic testing of the steam generators after plugging, and to maintain the other generators in wet lay-up. These operations require significant quantities of water. For example, filling and draining to locate leaks and to perform hydrostatic testing require 60,000 gallons of water per generator requiring tube plugging, plus 15,000 gallons for wet lay-up of the other two steam generators, for a total of approximately 75,000 gallons of water. An outage to perform the currently required periodic steam generator inspections consumes approximately 167,000 gallons of water. If tube plugging is required at the time of the inspection, an additional 40,000 gallons of water will be expended.

Following repair of the steam generators, it is expected that the steam generator tubes will remain intact; therefore, no unit shutdowns are anticipated for steam generator tube plugging and requirements for periodic inspection should be reduced significantly. 1

6.6.2 Operational Exposures

Section 3.3.7 discusses the reduction in man-rem exposure associated with repair. Due to the expected elimination of the necessity to plug steam generator tubes in the repaired steam generators, 14,000 man-rem may be saved over the lifetime of the plant.

6.6.3 Radiological Releases

Secondary plant activity results from primary to secondary leakage. The repaired steam generators will result in enhanced tube integrity thus reducing secondary plant releases.

6.7 REFERENCES

1. "Standards Publication for Gas Turbine Sound and its Reduction," NEMA SM 33-1964.

2. L. L. Beranek, "Noise and Vibration Control," McGraw Hill Book Company, 1971, Chapter 7, Pp 164-166.

7.0 EVALUATION OF ALTERNATIVES

7.1 INTRODUCTION

Repair, the replacement of the steam generator lower assemblies with new shop-fabricated lower assemblies, has been selected as the optimum solution at hand to correct denting. The discussion that follows demonstrates this. It also vividly indicates that the cost associated with the <u>outage</u> is the overriding consideration that <u>governs</u> any cost benefit evaluation.

The discussion that follows is based on the current state-of-the-art. It assumes that either one or both nuclear units must be shut down or that corrective action is required to ensure an acceptable level of system reliability. It must be noted that the technology as it relates to steam generator corrosion, electrical system requirements and economics are dynamic factors that directly impact the analyses provided below. At the shutdown of Unit 3 and/or Unit 4, evaluations will be updated as required to ensure that FPL embarks on the optimum approach to accommodate outage of the unit(s).

Loss of capacity from one or both Turkey Point units would require the addition of replacement capacity from new generating facilities and/or the purchase of firm power. The cost of new facilities can be compared with the cost of repair. However, the availability of firm power for purchase must be periodically re-evaluated to reflect current conditions. 1

Derating of one or both Turkey Point units is an alternative to the repair that cannot be addressed quantitatively at this time. Parametric studies can be performed assuming various derating conditions to determine the economics of repair versus derating. However, at this time corrosion rates and the likelihood of achieving a corrosion plateau cannot be quantified with precision. Accordingly, economic evaluations of derating do not presently provide a sufficiently reliable prediction of real world events. Should the evolving technology yield suitable corrosion models, further evaluation of derating would be warranted.

Potentially, there are several alternatives to the repair that could accommodate denting, viz., (1) arresting the corrosion phenomenon, (2) in-place restoration of dented tube areas (sleeving), and (3) in-place steam generator refurbishment (retubing). As discussed infra, the ability to sleeve is moot unless tube support plate (TSP) corrosion can be arrested. The ability to arrest TSP corrosion to ensure long term (30 to 40 year) operation without repair is not at hand.

In addition there are several potential means to accomplish the repair activity, viz:, steam generator removal from and entry to containment via (1) the equipment hatch, or (2) the containment wall, or (3) the containment dome.

The viability of each alternative to repair must be determined primarily by its present state of development. Alternatives that require research and development (R&D) to demonstrate feasibility are incompatible with the earliest potential shutdown date for initiation of repair activities, which is October-December 1978. 1

Table 7.1-1 is a summary comparison of the repair options. 2

7.4 IN-PLACE STEAM GENERATOR REFURBISHMENT

In principle, the methodology exists to refurbish the steam generators in-place. Although much of the technology exists, a comprehensive program of development and testing would be required to provide a basis for cost, time and personnel exposure comparisons.

To refurbish a steam generator, the upper or dome portion of the steam generator would be removed; the lower assembly internals and tubes would be removed and replaced with state-of-the-art internals and tubes; then the dome would be welded back in place. The in-place refurbished steam generator would be equivalent to the new shop-fabricated lower assemblies utilized in the repair effort.

Since both in-place refurbishment and repair result in equivalent steam generators, the viability of refurbishment must be based on economics, the availability of required tooling, and man-rem considerations. Refurbishment requires an R&D effort to develop (1) tools for retubing activities with a high enough production rate to reduce the unit outage to an economically acceptable level and (2) means to reduce personnel exposure to tolerable levels. This R&D effort is not compatible with the earliest potential outage date. 1

A comparative evaluation has been conducted to determine whether or not retubing could reduce the unit outage time to a point where it would be desirable to pursue the requisite R&D activities. This evaluation was based on the current state-of-the-art and assumed production rates for retubing activities believed to be achievable via a relatively short term R&D effort. The outage for refurbishment was estimated to be longer than the time required for replacement.

In-place refurbishment would require work within the steam generators to remove and replace tubing, etc. Shielding techniques and/or decontamination would be required to reduce radiation exposure to personnel. An R&D effort would be required to develop the means to reduce radiation fields to levels compatible with project manloading requirements. Even if exposure mitigating techniques are successfully developed, man-rem exposures associated with refurbishment are expected to exceed those associated with repair.

In summary, retubing does not currently offer an economic alternative to repair. If tooling that could appreciably reduce unit outage time were to become available, further consideration of this alternative would be warranted for efforts initiated sufficiently beyond the current October-December 1978 earliest potential start date to allow for completion of requisite R&D activities.

An evaluation of advantages/disadvantages and a cost breakdown estimate of in-place steam generator refurbishment are contained in comparison 2
Tables 7.4-1 and 7.4-2, respectively.

7.5 ALTERNATIVE REPAIR METHODS

Removal of the steam generators from containment and entry of the 1
shop-fabricated units into containment can be accomplished in one of three ways, viz.,

Scheme I

In this scheme, the steam generator lower assemblies would be removed and replaced through the existing equipment hatch. The upper assemblies (steam dome) remain inside the containment. Following installation of the new lower assemblies, the original upper assemblies will be welded to the new lower assemblies to complete the repair. An evaluation of advantages/ disadvantages and a cost breakdown estimate for Scheme I are contained in comparison Tables 7.4-1 and 7.4-2, respectively. 2

Scheme II

In this scheme, entire steam generators would be removed and replaced through a construction opening in the containment wall located above the equipment hatch, between elevations +58' and +98'. The opening would be about 20 feet wide and 40 feet high.

Scheme III

In this scheme, entire steam generators would be removed and replaced through a construction opening in the containment dome. The opening would be located above the steam generator closest to the equipment hatch. The opening would be about 20 feet in diameter.

Scheme III requires the use of an elevated rolling gantry on top of the containment. The gantry would move on two track beams (50 tons each). 1 Depending on the orientation of the rolling gantry, a postulated, though unlikely, collapse of the gantry and/or its rigging could impact safety-related plant functions. Thus, this scheme should appropriately be eliminated based on nuclear safety considerations. Schemes I and II do not result in any potentially unacceptable construction-related nuclear safety considerations.

Capital costs for repairing units 3 and 4, as well as a single unit's outage time (from shutdown to commencement of power operation) are as follows: 2

Alternative	Cost ($$10^6$)	Outage Duration (days)
Scheme I	102	207
Scheme II	116	340
Scheme III	107	300

Replacement power costs during the repair based on fuel cost differential are expected to be about $300,000/day/unit. The precise figure is not critical since it is obvious that a significant cost penalty is associated with each day of outage.

Based on the above, it is clear that Scheme I is the obvious repair alternative. Since there are no nuclear safety issues associated with Scheme I and other non-economic considerations, i.e., man-rem exposures are comparable for Schemes II and III, the unit downtime cost consideration dramatically favors this alternative.

7.6 MAN-REM CONSIDERATIONS

The preceding discussion demonstrates that repair appears to be the only long term method currently available to correct appreciable denting in steam generators. Of the three schemes potentially available to effect the repair, the man-rem exposure will be essentially the same since they differ only in the means of steam generator removal from containment. In-place refurbishment (retubing), although currently not a viable alternative, would likely involve a higher man-rem burden than the repair activity, based on today's state-of-the-art.

A specific dollar value to be assigned to the societal worth of a man-rem to the general public has been established on an interim basis by the Nuclear Regulatory Commission in Appendix I to 10 CFR 50. In its Statements of Considerations (40 XFR 19439), the Commission states with regard to cost-benefit requirements that "purely as an interim measure, we believe that we can accept the conservative value of $1,000 per total body man-rem for these cost-benefit evaluations. Since we realize that the ultimately accepted value may well prove to be less than this..."

The societal worth of a man-rem is set at $1000 for the discussion that follows. This value is utilized to quantitatively assess man-rem-related considerations as they relate to alternative comparisons presented in this report. The comparison is provided to comply with ALARA considerations in accordance with Regulatory Guide 8.8; however, this does not imply that comparisons of this nature are appropriate for man-rem evaluations. Compliance with 10 CFR 20 limits in conjunction with reasonable measures to minimize exposures, as discussed 1
in Section 3.3.5, provides an appropriate method of controlling occupational exposures.

Since the need for extensive steam generator inspection and tube plugging operations will be obviated by the repair, yearly exposures associated with these steam generator operations will be significantly reduced. The net result is that there will be a savings in man-rem over the life of the plant which is estimated to be 14,000 man-rem (see Section 3.3.7). Thus, based on the man-rem savings, the repair results in a $14,000,000 savings to society.

FPL subscribes to the precept of maintaining exposures ALARA. This principle must take into account the state of technology and the economics associated with any reduction in man-rem exposure. One overriding consideration is the duration of the unit outage. Since each day of unit unavailability is worth about $300,000, any man-rem reduction measures must result in savings of at least 300 man-rem per day of increased downtime. (The man-rem cost equivalent of the dose mitigating measures would obviously have to be added to the 300 man-rem per day value.)

Potential exposures associated with the repair program are about equally divided between repair and disposal. About 50 percent of

the man-rem burden can be eliminated by storing the existing steam generator lower assemblies onsite and decommissioning them with the plant. Since disposal subsequent to removal can result in 750 - 1500 man-rem per plant, long term onsite storage could result in a man-rem savings as high as 3000 man-rem. The corresponding societal savings would be worth in the order of $3,000,000.

Exposure associated with removal and replacement of the steam generators is due to many activities occurring in radiation fields generated by radioactivity within the steam generators and the remainder of the reactor coolant system. Chemical decontamination of the steam generators would reduce these exposures. However, the decontamination activity itself would result in additional exposure of personnel and large quantities of liquid waste. The net savings in man-rem is not expected to exceed 75 - 200 man-rem per plant. (It must be noted that the man-rem savings associated with decontamination could be negligible, or possibly negative, i.e., increased man-rem could result from in-place decontamination. See Section 8.2 for a discussion of decontainmination.)

Steam generator decontamination would increase the unit outage by about 14 days and would have a substantial cost associated with it. Based on the outage alone, a man-rem savings of 4200 man-rem per unit would be required to justify in-situ decontamination. The savings possible are more than an order of magnitude less than that required to compensate for the increased outage. 1

In summary, man-rem evaluations indicate that (1) a potential societal savings of $17,000,000 results from repair and onsite storage with disposition during decommissioning and (2) steam generator removal would result in personnel exposures being as low as is reasonably achievable, without in-place decontamination.

7.7 REPLACEMENT CAPACITY

If FPL is required to permanently shut down one or both Turkey Point nuclear units it would have to replace this capacity to ensure adequate electrical system reliability. Assuming action was initiated in the near future to construct replacement capacity, it would be physically impossible to construct new combustion turbines to replace the nuclear units before 1981, new coal units before 1985, and nuclear units before 1988. The nuclear units at Turkey Point are used for base load operation. Combustion turbines are used to supply peaking power, and thus are not suitable as replacement capacity for these units due to their high fuel and operating costs. In addition, the use of combustion turbines as replacement capacity would increase the use of petroleum products, which is contrary to what we perceive to be the National Energy Policy.

Currently firm purchase power is not expected to be available in quantity after 1981. The availability of firm purchase power will be reevaluated as required to ensure current conditions are reflected.

The repairs to the steam generators at Turkey Point are justified when the associated costs of repair are compared to the cost to build replacement capacity. The steam generator repairs are estimated to cost 77 $/kW which must be compared to the costs for replacement capacity provided below. This comparison establishes repair as the current cost-effective choice for satisfying FPL's electrical system requirements.

TYPE OF UNIT	COST ($/kW)	INITIAL OPERATION
Gas Turbine	224	1981
Coal	1059	1985
Nuclear	1448	1988

7.8 DERATION

Section 7.2 discusses TSP corrosion. It indicates that it is not currently possible to predict whether or not a corrosion saturation level or plateau will occur. Should such a plateau become predictable, it would be possible to define a power condition associated with the corrosion plateau. Economic studies could then be conducted to assess the propriety of repair.

Section 7.2 also addresses inhibitors. Should these be successfully developed, rates of TSP corrosion may be reduced dramatically. This could allow for a ramped or linear rate of possible deration over many years. Economic studies could be conducted to determine whether ramped deration or repair is the optimum choice. Alternately, repair might be delayed until a more attractive date. 1

Unfortunately, inhibitors and the ability to predict a TSP corrosion plateau with precision are not at hand. Thus, it is not possible to conduct meaningful deration studies at this time.

7.9 CONCLUSIONS

Repair of the steam generators via the containment equipment hatch pathway is the preferred choice. The dramatic economic advantage due to reduced outage time offsets any potential advantages associated with other viable containment removal schemes.

The ability to arrest TSP corrosion at Turkey Point is not within today's state-of-the-art. Sleeving does not offer any potential benefit without the ability to arrest TSP corrosion. In-place refurbishment (retubing) requires R&D to develop the tooling necessary to make this alternative economically competitive and R&D to develop means to reduce man-rem exposures to acceptable levels. There is currently no suitable alternative to the repair of the Turkey Point steam generators.

There are two principal societal considerations associated with repair, viz., the duration of the unit outage and man-rem exposure. Repair of a unit could involve an upper limit exposure approaching 1450 man-rem, which has a worth

of about $1,450,000. A 207-day outage at about a $300,000/day replacement
power cost has a worth of about $62,100,000 or about 43 times that associated
with the man-rem worth. Clearly then, any emphasis for reducing societal
costs should be focused on reducing unit unavailability. This is reinforced
due to the fact that man-rem associated with repair will be offset by a sub- 1
stantial reduction in operating man-rem subsequent to repair with a net man-rem
societal savings worth $14,000,000 over the lifetime of both Turkey Points
units.

Onsite storage and decommissioning of the existing steam generators meet the guidelines of the ALARA principle. Chemical decontamination of the steam generators does not meet the guidelines of this principle. (See Section 8.0 for a more detailed discussion of storage and decontamination.)

TABLE 7.1-1

COMPARISON OF THE REPAIR OPTIONS

	Repair Steam Generators	In-place Refurbishment	Continuing Operation in Present Mode	Continuing Operation with Corrosion Inhibitor	Derating	Replacement with Fossil Units	Replacement with New Nuclear Units
Comparative Capital Costs	77$/kW	88$/kW	A dollar value has not been estimated for this option.	The state-of-the-art for corrosion inhibitors must be further developed by technical research before meaningful evaluations can be made with respect to effects on cost, reliability, reserve margin, and public health and safety.	A dollar value has not been estimated for this option.	1059 $/kW - Coal 224 $/kW - Gas Turbine (The installation of gas turbines would increase the nation's reliance on petroleum products, which we perceive to be contrary to the National Energy Policy.)	1448 $/kW
Effect on Reliability	Following the repair, the steam generators will have state-of-the-art internals and tubes. Reliability, therefore, would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.	Following the refurbishment, the steam generator will have state-of-the-art internals and tubes. Reliability, therefore, would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.	Outages will be required for inspecting and plugging steam generator tubes. It is conceivable that continuation of the corrosion mechanism could eventually result in unacceptable inspection intervals and reductions in unit power.		Even though derating occurs, outages will be required for inspecting and plugging steam generator tubes.	The reliability of large coal units is comparable to that of nuclear units of similar size. Other units, such as combustion turbines, are not considered suitable for base load service.	Reliability would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.
Effect on Reserve Margin	Reserve margin would be reduced during the outage period.	Reserve margin would be reduced during the outage period.	Continued steam generator corrosion could result in reduction in unit power or, possibly, total unit shutdown. Should either occur, system generating capacity reserves would be reduced and replacement power would eventually be required. (Currently, firm purchase power is not expected to be available in necessary quantity after 1981.)		As the units are derated and possibly, eventually, shut down, system generating capacity reserves would be reduced and replacement power would eventually be required. (Currently, firm purchase power is not expected to be available in necessary quantity after 1981.)	Reserve margin could be reduced between the time that the nuclear units would be derated or shut down and the replacement fossil units are fully operational.	Reserve margin could be reduced and replacement power may be required between the time that the nuclear units would be derated and possibly, eventually, shut down and the replacement nuclear units are fully operational. (Currently, firm purchase power is not expected to be available in quantity after 1981.)
Effect on Public Health and Safety	During and following the repair, radioactive releases should be less than experienced in 1976 at Turkey Point Units 3 and 4. Furthermore, a net savings of approximately 14,000 man-rem is expected by implementing the repair.	Since developmental work is required for this option, it is not possible at this time to accurately predict radioactive releases during the outage. However, it is expected that radioactive releases and man-rem doses over the life of the plant will be reduced by implementing refurbishment.	The inspection and plugging criteria for steam generator tubes and the reduction in unit power, if required, are established to ensure that there are no undue risks to the public. While some small amount of radioactivity is released as a result of primary to secondary leakage, all plant parameters and radioactive releases will continue to be within Technical Specification limits.		The inspection and plugging criteria for steam generator tubes and the reduction in unit power are established to ensure that there are no undue risks to the public. While some small amount of radioactivity is released as a result of primary to secondary leakage, all plant parameters and radioactive releases will continue to be within Technical Specification limits.	There are some public health and safety effects associated with fossil power plants due mainly to the release to atmosphere of combustion exhaust products.	With new nuclear units, radioactive releases and man-rem doses will be reduced significantly.

.TABLE 7.4-1

COMPARISON OF IN-PLACE
STEAM GENERATOR REFURBISHMENT TO STEAM GENERATOR
LOWER ASSEMBLY REPLACEMENT

	In-place Refurbishment	Lower Assembly Replacement
Advantages	1. No reactor coolant system pipe cuts are required.	1. The repair can be accomplished using existing technology.
	2. Less containment internal concrete must be removed/ replaced when compared to lower assembly replacement.	2. Lower man-rem doses are expected than for in-place refurbishment.*
	3. Rigging and transport loads are lighter than those for lower assembly replacement.	3. Lower cost in incurred than for in-place refurbishment.*
	4. The repaired steam generators will have state-of-the-art internals and tubes.	4. Shorter outage is required than for inplace refurbishment.*
		5. Radioactive steam generator lower assemblies may be easily sealed and stored as a unit thereby eliminating the cutup, packaging, and shipping of radioactive steam generator materials.
		6. Lower airborne activity levels may occur since no cutting of contaminated tubes is required.*
		7. The repaired steam generators will have state-of-the-art internals and tubes. 2
Disadvantages	1. R&D is required for retubing tools and techniques that yield acceptable production levels.	1. Reactor coolant system pipe cuts are required.
	2. Higher man-rem doses could be realized than for lower assembly replacement (even after development of exposure mitigating techniques).*	2. More containment internal concrete must be removed/ replaced when compared to in-place refurbishment.
	3. Higher cost is incurred than for lower assembly replacement.*	3. Rigging and transport loads are heavier than those for in-place refurbishment.
	4. Longer outage is required than for lower assembly replacement.*	
	5. Considerable difficulty is associated with handling, storage and ultimate disposition of the individual contaminated tubes.*	
	6. Higher airborne activity levels may occur due to the cutting of the individual contaminated tubes.*	

* These advantages/disadvantages are based on the current state-of-the-art. Research and development activities may eventually eliminate these current advantages/disadvantages.

TABLE 7.4-2

ESTIMATED CAPITAL COSTS ASSOCIATED WITH IN-PLACE STEAM GENERATOR REFURBISHMENT AND LOWER ASSEMBLY REPLACEMENT

	Item	Refurbishment* Costs (dollars)	Repair Costs (dollars)
1.	Costs of Steam Generator Equipment	$ 25,400,000	$29,000,000
2.	Development Program for Retubing	17,125,000**	4,300,000
3.	Engineering Costs	8,400,000	7,300,000
4.	Field Costs	27,500,000	31,500,000
5.	Power Resources Costs	1,600,000	1,600,000
6.	FPL Support Costs	3,000,000	3,000,000
7.	Construction and Engineering Fees	1,400,000	1,500,000
8.	Allowance for funds used during construction	13,300,000	9,000,000
9.	Contingency	18,350,000	14,800,000
	TOTAL	$116,075,000	$102,000,000

*Scoping estimate based on current available technology.

**This amount also includes field use of tooling and technical support of retubing.

8.0 COST BENEFIT ANALYSIS FOR THE REMOVAL, STORAGE AND DISPOSITION OF THE LOWER ASSEMBLIES CONSIDERING ALARA

8.1 INTRODUCTION

Steam generator lower assembly removal, storage, and disposal were evaluated. The cost of each was determined and compared with benefits gained in the total man-rem reduction to workers in accordance with the philosophy of reducing worker dose to levels which are ALARA. The following are evaluated herein: (a) In-place decontamination, and (b) steam generator storage and disposal methods.

a. The potential benefits of in-place decontamination of the steam generators must be derived from the increased cost for the decontamination plus the increased outage time, weighed against the net savings in man-rem as well as any potential gain in labor productivity.

b. The optimum method for steam generator storage and/or disposal
must be derived from the cost associated with the storage
facility, activities related to preparation of the lower
assemblies for shipment, e.g., possible decontamination, cut
up, packaging, etc., shipment to a disposal site, and burial, 1
weighed primarily against the net savings in man-rem.

8.2 STEAM GENERATOR IN-PLACE DECONTAMINATION

If the steam generators were to be chemically decontaminated prior to removal, the reduction in worker dose should offset the cost, schedule, licensing, and other considerations associated with the decontamination activities. Although chemical decontamination of commercial reactor systems has been accomplished successfully (e.g., Shippingport), the state-of-the-art for the process has not convincingly demonstrated that the benefits outweigh the problems and costs. For this study, it was assumed that the steam generators will be isolated from the rest of the reactor coolant system for decontamination by welding temporary plates or installing temporary mechanical seals inside the hot and cold legs.

The scheduled outage time is impacted if in-place decontamination is implemented due to the required preparation of the reactor coolant system and the decontamination activites themselves, which must be completed prior to cutting of the reactor coolant piping. An additional schedule penalty not assessed here may be incurred due to interruption of work activities within containment due to the expected high radiation fields during circulation of the decontamination solutions.

A schedule extension of at least 2 weeks is necessary to accommodate decontamination of the three steam generators (in-place decontamination would be on the critical path). Many uncertainties are associated with decontamination, e.g., insertion of the seals in the reactor coolant lines, decontamination effectiveness, etc. However, for conservatism, no schedule penalty was assigned to these uncertainties.

A man-rem dose assessment was made for the repair effort considering in-place decontamination of the steam generators. The results of this assessment are included in the cost-benefit analysis presented below. This dose assessment was done based on the radiation surveys and the lower assemblies removal/ installation tasks presented in Subsection 3.3.7.2. In addition, a decontamination factor of 10 for the average reduction in working fields was assumed for the primary side surfaces of the steam generators, although actual results could vary considerably, either higher or lower.

Assuming in-place decontamination of the steam generators prior to initiation of the repair effort, it is estimated that approximately 200-400 man-rem per unit could be incurred during the removal of the lower assemblies. In addition, it is estimated that approximately 25-100 man-rem per unit could be incurred during the decontamination effort itself, including processing and solidifying of decontamination wastes.

Some activities associated with in-place decontamination may result in significantly greater man-rem accumulation and costs than assessed herein, if unforeseen difficulties are experienced. For example: the design and placement of the RCS leg seals require an R&D effort before an accurate assessment of the man-rem exposure associated with the installation of such seals can be made. To illustrate this, the following examples were chosen for the sole purpose of assessing the man-rem associated with such an effort. 1

a. One method of sealing the RCS piping is to weld a steel plate into the nozzles of the steam generators. The seal plate could be of steel and fabricated in two pieces in order to get the approximately 31-inch diameter seal plate into the 16-inch manway in the channel head of the steam generator. Once through the manway, the semi-circular plates would be welded together and then welded to the interior of the nozzle.

   It is estimated that such an effort could require more than one shift to install each seal inside the channel head and that 1200-4000 man-rem could be accumulated during the installation of six plugs within three steam generators.

b. A second method of sealing the RCS piping involves the use of a modified quick-sealing piping system pressure test plug. This method would consist of fabricating a standard quick-sealing test plug in pieces, so that they fit through the 16-inch manway in the steam generator channel head. These quick-sealing plugs consist of steel backing plates to withstand the forces induced by the water pressure and seals compatible with the decontamination fluid to prevent leakage.

   It is estimated that two men would be required inside the channel head to install the plug and two men outside to move the plug pieces into the channel head through the manway. It is further estimated that it would take between

8 and 16 minutes to install each plug, including the time spent in getting into and out of the channel head and that 30-60 man-rem could be accumulated during the installation of six plugs in three steam generators.

The man-rem ranges cited for sealing RCS legs are considered the extremes; however, it is obvious that such a task could induce a significant increase in the worker man-rem burden. This additional man-rem burden is not included in net decontamination man-rem savings estimates.

NRC review and approval of the chemical decontamination process may result in additional cost and schedule impact. This review and approval process could also delay the earliest potential repair effort initiation date of October-December 1978.

Based on the information provided above and in Subsection 3.3.7.2 and excluding items such as RCS leg seal installation, it is estimated that approximately 150-400 man-rem (two-unit total) could be saved by implementing in-place decontamination of the steam generators in addition to the ALARA considerations presented in Section 3.3.5.

The costs, schedule impact, and man-rem savings are presented below in the cost-benefit analysis. It is obvious that the combined effects of schedule penalty and the large capital cost associated with decontamination far outweigh the 1
maximum potential benefit in man-rem savings. It is therefore concluded that in-place decontamination of the steam generators is not ALARA.

	Units 3 and 4
Decontamination Capital Cost	$ 2,000,000
Schedule Penalty (2 weeks - 2 units)	$ 8,400,000
Total Evaluated Cost	$10,400,000
Man-rem savings	150-400

8.3 STEAM GENERATOR STORAGE AND DISPOSAL

The steam generator lower assemblies will be stored onsite, at least temporarily, prior to eventual disposal. The alternatives of storage and disposal of the steam generator lower assemblies following their removal from the containment are addressed in this section. The evaluation was divided into two basic categories: (1) method and time period for storage, and (2) preparations and methods for disposal. It should be noted that a cost benefit analysis for the disposal technique as discussed in Subsection 3.4.3.2, i.e., barge shipment of the entire lower assemblies, has not been performed since barge receipt capabilities are currently not available near a licensed burial facility. The following discussions therefore include the costs associated with a temporary storage facility.

8.3.1 Lower Assembly Storage

Two basic storage concepts were evaluated: (1) storage until the end of the plant lifetime then disposition during decommissioning, and (2) interim short term storage prior to cut up and shipment to a burial site.

8.3.1.1 Long Term Lower Assembly Storage

As discussed in Section 3.4.6, the Units 3 and 4 lower assemblies will be sealed prior to storage. This results in a complete encapsulation of residual contamination inside the steam generator. Release of airborne or liquid radioactive materials will not occur; therefore, enclosed storage is not required (i.e., floor and roof need not be provided). However, shielding would be required around the assemblies and access control measures would be provided during the storage period.

At the end of the plant lifetime, disposition of the steam generators will be accomplished in conjunction with plant decommissioning.

8.3.1.2 Lower Assembly Storage with Provisions for Interim Short Term Cut Up and Shipment

If the lower assemblies are to be cut up for shipment, additional contamination control measures, as well as shielding, would have to be employed as discussed in Subsection 3.4.3.1. For this option, an enclosure with appropriate controls for airborne and liquid effluents will be required in addition to the requirements set forth in Section 3.4.2.

8.3.2 Cutup Preparation and Method of Disposition

The lower assemblies will be stored and prepared for disposition by one of the 1
methods discussed in Sections 3.4.2 or 3.4.3.

8.3.2.1 Near Term Cut Up and Disposal Without Decontamination

Man-rem burdens associated with cut up and disposal of the lower assemblies without decontamination, as shown in Section 3.4.4, are relatively high. Techniques for field cut up and packaging are not well established which makes accurate cost prediction uncertain.

8.3.2.2 Near Term Cut Up and Disposal with Injection of Solidification Agent

For the purposes of cut up, the contamination encapsulated in the lower assemblies can be "fixed" during cutting operations by injecting a solidification agent. This is as an alternate to decontamination of the lower assemblies in storage.

The lower assemblies would be prepared by cutting the nozzle closure plates and welding inlet and outlet connections prior to the injection of the solidification materials. The materials would be pumped into the steam generator channel head and tubes and allowed to solidify.

Injecting the solidification agents into a complex geometry such as the steam generator and having assurance that most of the 3260 tubes will get filled has a large uncertainty at this time. A research and development (R&D) program would be required to define materials, methods and equipment requirements to achieve the desired results. Without this R&D program, precise cost and man-rem predictions are difficult to assess.

8.3.2.3 Near Term Cut Up and Disposal with Chemical Decontamination

Decontamination of the lower assemblies while in storage can be done similar to the in place decontamination discussed in Section 8.2. However, it can be performed with less cost and less worker dose in storage since more space is available to set up equipment, shielding, etc. Additionally, the task can be deferred to take credit for the additional decay time of radioisotopes.

As with the case of solidification, only partial success may be achieved. This is due to the fact that some of the tubes are already plugged. Difficulty is also expected in wetting all of the tube surface area since it may not be possible to force all of the air out of the tubes.

The cost estimate includes the decontamination process plus the cut up, packaging, shipment and disposal. It was assumed that the fewest number of cuts would be made which would meet size, weight and radioactive shipping restrictions. Torches or cutting machines would be employed in a semi-automatic manner as appropriate to minimize worker dose.

8.3.2.4 Long Term Storage with Deferred Cut Up and Disposal

The lower assemblies can be stored sealed, in open storage until the end of the plant lifetime. At that time the radioactivity will be less than 1 percent of the levels expected when the lower assemblies are first placed in storage. Cut up and disposal can be accomplished by one of the methods discussed above. At that time, temporary contamination control measures, as discussed in Subsection 3.4.3.1, would be employed. 1

8.3.2.5 Lower Assembly Disposition During Decommissioning

If the lower assemblies are stored until the plant is decommissioned, radioactive contamination will be less than 1 percent of the present levels. Disposal during decommissioning will therefore result in lower man-rem and lower costs for contamination control and disposal efforts. The costs associated with this option include the surveillance program outlined in Section 3.4.6. Decommissioning costs approximate those in the AIF decommissioning study. (Reference 1)

It should be noted that at some future time it may become a viable, competitive option to ship the lower assemblies intact to an offsite burial facility. Since there are no current facilities which can accommodate this option, a reliable cost estimate cannot be made at this time.

8.3.2.6 Conclusions

The cost and man-rem estimate (see Section 3.4.4) for each method of lower assembly disposition are summarized below. It is seen that the lowest cost and the lowest man-rem burden is associated with the long term, onsite storage and decommissioning of the lower assemblies.

		Cost	Man-Rem
a.	Cut up and disposal near term with no decontamination	$4,560,000	1500-3000
b.	Cut up and disposal near term with solidification agent	$4,220,000	800-1600
c.	Cut up and disposal near term with decontamination	$4,750,000	250-1100
d.	Long term storage with deferred cut up and disposal	$2,490,000	20-40
e.	Long term storage with disposition during decommissioning	$2,020,000	1-3

As indicated in Section 3.4.4, the occupational exposure associated with on-site storage of six steam generator lower assemblies, without decontamination, for a period of 35 years and subsequent disposition, is less than 40 man-rem. 1
On the other hand, the exposure associated with near-term disposal is greater than 250 man-rem. Therefore, it is readily apparent that near-term disposal of the lower assemblies is not cost beneficial when compared with long-term storage and in fact is detrimental from a man-rem exposure standpoint.

Onsite storage and decommissioning would result in personnel exposures being as low as is reasonably achievable.

8.3.3 <u>References</u>

1. AIF Decommissioning Study, AIF/NESP-009, November 1976, Chapter 3 and Appendix A.

8.4 SINGLE UNIT CONSIDERATIONS

The discussions above are based upon repair of both nuclear units at Turkey Point. For the reasons cited below, the conclusions reached remain valid if only one unit is repaired.

Costs per steam generator will be greater if only one unit is repaired since fixed costs associated with a given operation are spread over three rather than six steam generators. For example, equipment to process decontamination fluid must be installed whether or not three or six steam generators are decontaminated.

2. The ramp fill collapses causing the loaded transporter to tilt.

3. The lower assembly drops from the transporter at the top of the ramp then rolls off the edge of the ramp and impacts the structure.

Results of these analyses indicate that the analyzed structures are capable of withstanding the impact loads without loss of integrity.

Postulated Loaded Transporter Brake Failure

Analyses were performed to determine the capability of the radwaste building to withstand the impact caused by a postulated brake failure of a loaded transporter traveling down the ramp. Due to the orientation of the transporter at the equipment hatch, neither the Unit 3 auxiliary building nor the diesel generator building could be impacted by this postulated incident.

The following assumptions were made for the purpose of these analyses:

1. The brakes fail on a loaded transporter at the top of the ramp.

2. The loaded transporter travels from the top of the ramp to the bottom of the ramp prior to impact.

3. The loose fill embankment extending the full length of the radwaste building, as shown in Figure A.6-2, will be impacted prior to impact with the radwaste building wall.

Results of these analyses indicate that the radwaste building is capable of withstanding the impact loads, which are distributed through the embankments to the walls, without loss of integrity.

b. The Unit 4 removed lower assemblies will be transported south on the containment access road, around the southern end of the radwaste building, then east along the road south of the ash disposal pits. The lower assemblies will proceed down a ramp south of the pits to the temporary onsite storage facility which will be located in the laydown area shown on Figure 3.1-2.

The Unit 3 removed lower assemblies will be transported northeast on the containment access road, south along the east plant road, east along the road south of the ash disposal pits, then down a ramp to the storage facility.

Should it be decided at some future time to transport the removed lower assemblies (either intact or cut up) to a land burial site, the following routes may be utilized:

1. Transport from the storage facility up a ramp, west along the road south of the ash disposal pits, then north along

the east plant road to a point north of Unit 1. At this point the lower assemblies could be loaded on barges in the turning basin or transported off site through the north gate.

2. Transport from the storage facility up a ramp, west along the roads south of the ash disposal pits and south of Unit 4, across the discharge canal bridge, then offsite. This route does not require transport of the lower assemblies over any safety related piping or duct banks.

c. There are two types of hauling devices under consideration for steam generator transport:

1. A flat-deck, multi-axle, 250 ton-capacity trailer with jeep dolly. The trailer would be supported by 4 rear axles with 8 tires each, 2 jeep axles with 8 tires each and 2 prime mover axles with 4 tires each. The bed would measure approximately 12' X 42'.

2. A 250 ton-capacity dolly system utilizing steel framing for lower-assembly support. The framing would be supported at each corner by a 16-wheel dolly set; the prime mover would be separate. The dolly set gauge would be adjustable to provide acceptable bearing pressure.

d. As discussed in the response to Question 1 of this Appendix, the results of analyses summarized in Subsection 5.2.1.2 are applicable to all crane incidents including those caused by operator malfunction.

e. As discussed in the response to Question 1 of this Appendix, the results of the analyses summarized in Subsection 5.2.1.2 are applicable to all crane incidents including those caused by a crane tipping with a lower assembly at its highest lift position.

8. Provide information to ensure that the on-site quality assurance requirements for repair related components that are part of the primary reactor coolant pressure boundary will meet the appropriate recommendations of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

RESPONSE

FPL and Bechtel comply with the intent of Regulatory Guide 1.37 with minor deviations as stated in the "FPL Topical Quality Assurance Report" (FPLTQAR 1-76A), Revision 2, September 8, 1977, and the "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, August 15, 1977, respectively.

13. Regulatory Guide 1.38 "QA Requirements for Packaging, Shipping, Receiving, Storage, and Handling for Nuclear Power Plants," has not been included in the list of applicable regulatory guides specified in paragraph 2.1.4. And, while not yet endorsed by a regulatory guide (but acceptable to NRC), the Requirements for Auditing of QA Programs for Nuclear Power Plants, ANSI N45.2.12-Draft 3, Revision 4, February 22, 1974, have also not been included in the list. Please include these in the list or provide information as to why they should not be included.

RESPONSE

FPL and Bechtel comply with the intent of Regulatory Guide 1.38 and ANSI N45.2.12 with minor deviations as stated in the "FPL Topical Quality Assurance Report," (FPLTQAR 1-76A), Revision 2, September 8, 1977, and the "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, August 15, 1977, respectively

The Westinghouse position on Regulatory Guide 1.38 presented in revised Section 2.1.4 of this report. The Westinghouse position on ANSI N45.2.12 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

19. Indicate what actions you plan to take to minimize condenser leakage or to improve the performance of your condenser.

RESPONSE

The main condensers at Turkey Point, Units 3 & 4, are progressively being retubed with titanium tubing. Based upon industry and FPL experience, FPL believes that titanium tubing is the best choice of tubing material currently available for approaching a leak-free condenser.

The data below show the retubing efforts performed at Turkey Point.

Unit	Quadrant	Date in Service	Tubing Material
Unit 4	4BS	June 1976	70/30 Cu-Ni
Unit 4	4BN	June 1976	Titanium
Unit 3	3AN	Jan. 1977	Titanium
Unit 3	3AS	Feb. 1978	Titanium

With respect to Unit 4, those quadrants not previously retubed are planned for a retubing with titanium during the steam generator repair.

An Amertap on-line condenser cleaning system has been recently installed at Unit 3 (in service 2/78). This system is expected to improve condenser thermal performance and reduce tube corrosion by eliminating calcium scale buildup. Retention of the Amertap system on Unit 3 and potential installation of an identical unit on Unit 4 will depend upon the experience observed at Unit 3.

20. Indicate your plans, if any, to make modifications to reduce the amount of copper (or other metal ions) in the condensers and heaters in feedwater and condensate system in view of the Westinghouse findings on the role of Cu^{++}(or other metal ions) on denting.

RESPONSE

Investigation of the denting phenomena has shown that the denting was caused by the production of a magnetite (Fe_3O_4) corrosion product in the tube/tube support plate annulus of the currently employed carbon steel support plates. Lab tests indicate that Cu ions enhance the carbon steel corrosion rate.

However, the replacement steam generators are being built with SA-240 type 405 SS tube support plates employing a quatre-foil design. Corrosion tests and careful metallographic examinations have confirmed that SA-240 Type 405 SS in metallic chloride environments forms a thin, tight, adherent corrosion film with little change in overall specimen dimension. The test results to date indicate that SA-240 Type 405 SS has exhibited no denting under identical test conditions where carbon steel has produced significant denting.

It should be noted that the following will reduce the amount of copper and metal ions within the steam generators:

1. An improved internal blowdown design and improved steam generator thermal hydraulics have been incorporated into the replacement steam generator and are expected to improve the efficiency for blowdown removal of sludge and corrosion products.

2. Additional access ports will be provided to enhance the sludge lancing capability of the replacement steam generators.

3. Planned condenser modifications as provided in the response to Question 19 will significantly reduce copper bearing materials in the condensate systems.

27. If decontamination is necessary, demonstrate that the methods and the decontamination solutions will not degrade or adversely affect the reactor coolant piping or components which are part of the primary system pressure boundary. Further show that the decontamination solutions will not have deleterious latent effects in subsequent plant operations.

RESPONSE

In-place chemical decontamination of the reactor coolant piping or components which are part of the reactor coolant pressure boundary is not planned for the steam generator repair effort.

As indicated in Subsection 3.3.5.3 and in the response to Question 40 (c) of this Appendix, local/spot decontamination of the steam generators and reactor coolant piping will be performed as required. Local decontamination will consist of cleaning of loose surface contamination with either soap and water or RADIAC solutions. Past industry experience has shown that these standard solutions are not deleterious to the piping or components involved.

41. Section 3.3.5.4 stated that "low background radiation waiting areas will be established." Provide the criteria to be used in establishing these areas. Describe how the airborne radioactive contaminants will be controlled in these areas.

RESPONSE

Low background radiation waiting areas will be established using the criteria that the radiation field within the area will be no greater than Zone III ($\leq$15 mr/hr). These waiting areas will be established by the Turkey Point Health Physics staff based on radiation surveys taken periodically by Health Physics personnel throughout the repair effort.

As described in Section 3.3.3, airborne radioactivity inside containment during the repair effort will be controlled, monitored, and released via the plant vent stack by operating the containment purge system. In addition, radioactivity generated during the cutting of the reactor coolant pipes will be contained within specially designed contamination control envelopes, which will provide high efficiency filtration. These measures will maintain airborne radioactivity concentrations at acceptably low levels in the waiting areas.

7

46. Section 3.4.2 states that a temporary onsite storage area will be provided for the storage of the steam generator lower assemblies. Details of the storage area, including its location, design bases, and radioactivity control systems, should be presented. How long will the steam generators remain at the temporary storage facility? Evaluate the effect of decay time (e.g., 0, 1, 5, 10, 25 years) of the radioactivity in the steam generators relative to the selection of the time of ultimate disposal of the steam generators and man-rem exposures. Clearly indicate how and when the steam generators will be disposed of.

RESPONSE

It is anticipated that the temporary onsite steam generator storage facility will be located in the laydown area south of the ash disposal pits shown on Figure 3.1-2. The facility is described in Section 3.4.2.

As described in Section 8.3, an evaluation was performed to determine the most cost-beneficial method for disposition of the removed lower assemblies. It was concluded in Subsection 8.3.2.6 that the lowest cost and the lowest man-rem burden is associated with long-term, onsite storage and disposal during decommissioning.

Figure A.46-1 is a typical decay curve of percent steam generator gamma activity versus time following reactor shutdown. The initial activities used to generate the decay curve are given in Table 5.2-1. With this decay curve, the effect of lower assembly radioactive decay can be directly related to the time of ultimate steam generator disposal and to the associated man-rem exposures.

49. For the entire steam generator replacement program, estimate the total volume and curies of solid wastes which are to be disposed of including transuranics, Iodine 129, Iron 55 and Nickel 63.

RESPONSE

The major source of solid radioactive waste associated with the repair effort is the steam generator lower assemblies. Each lower assembly has an approximate volume of 3500 ft^3 and an approximate radioactivity content of 250 curies of deposited gamma activity, as stated in Section 3.4.

As discussed in Subsection 3.3.6.1, approximately 60 cubic yards of concrete per unit will be disposed of during the repair effort. It is expected that the radioactivity content in this concrete will be only a fraction of a curie.

Reactor coolant pipe waste generated during the cutting and weld end prep operations will be disposed of as radioactive material. It is estimated that these shavings may amount to approximately 7 cubic feet and contain approximately 1 curie of radioactivity per unit.

There will also be some additional low-level radioactive materials. The quantities involved are expected to be the same order of magnitude as the quantities disposed of following a refueling outage. Typical quantities and types of radioactive wastes for a refueling outage are as follows:

	Waste Material	Quantity (ft^3)	Activity (Ci)
a.	Rags, paper and clothing	22,000	~50
b.	Sand, concrete, tools, and scaffolding (packaged as is)	3,000	~50
c.	Evaporator bottoms	700	~ 5*
d.	Spent resin	100	~25*

*These values were obtained by utilizing typical radioactive concentrations per unit of waste volume from Table 2-4, A.E. Ackens, Jr., et al., "Migration of Radionuclides at Low-Level Waste Burial Grounds," Dames & Moore Report for The AIF, December 1977 (in press).

2. ELECTRICAL

Identify the cable circuit inside the containment connected to the steam generator and any other system associated with the repair of the steam generator, and provide the safety analysis to prove that no adverse interaction occurs to the operating unit.

RESPONSE

As discussed in Section 3.2, detailed engineering studies are in process to precisely define the components, pipes, cables, instruments, etc. within the containment affected by the repair activity. The results of these studies to date are contained in Sections 3.2.1 through 3.2.5 and, for instrumentation and electrical equipment, are summarized below.

a. Unit 3 and Unit 4 mechanical equipment to be temporarily relocated is listed in Section 3.2.1. The disconnection and reconnection of power cables to the equipment is discussed in Section 3.2.3. The associated feeder breakers for the equipment will be opened and tagged.

b. Unit 4 instrumentation to be temporarily removed is listed in Section 3.2.2; Unit 3 instrumentation to be removed is still under study. The removal and reinstallation of the associated cables is discussed in Section 3.2.3.

c. Unit 3 and Unit 4 conduit to be temporarily removed is listed in Table 3.2-1. The removal and reinstallation is discussed in Section 3.2.3.

As discussed in the response to Question 3, Appendix A, no postulated rigging incident inside the containment of the shutdown unit could adversely affect the safety-related functions of the operating unit.

a. Contamination control envelopes for use during cutting operations will be constructed of fire retardant materials.

b. Combustible materials will be minimized in areas in which cutting and welding operations are being conducted.

c. Combustible materials inside the contamination control envelopes will be minimized.

d. A fire watch will be present during welding and cutting operations.

e. Adequate fire extinguishers are available throughout the containment and portable fire extinguishers will be readily accessible in the work areas when cutting and welding is performed.*

f. Fire hose of sufficient length to reach the most remote steam generator compartments as well as containment areas will be available and dedicated to fight a postulated fire inside the containment.

3. In addition to the fire extinguishers already in the containment, portable fire extinguishers will be readily accessible in the work area as necessary, in accordance with existing FPL welding and cutting procedures.*

4. During the repair, the existing containment lighting system which includes dc emergency lighting will be operable.

5. Procedures for handling used protective clothing during a normal refueling shutdown will be employed during the steam generator repair. These procedures establish personnel access and egress so that the personnel undressing areas are located outside of the containment, in outside open areas, thereby obviating the need for covered noncombustible containers for disposing of used protective clothing.

6. Normal refueling shutdown practices for plastic sheeting and paper products will be employed during the steam generator repair. Also, controls will be established for the removal of combustibles as stated in item 8.*

7. A portable foam system, suitable for use on hydrocarbon liquid fires, will be available on-site during the repair.

8. All administrative site procedures will be reviewed for the control of combustibles. Controls will be established for the removal of all combustible waste, debris, scrap, oil spills, or other combustibles resulting from the work activity (unless suitably contained) in the area following completion of the activity or once per 24 hours, whichever is sooner.*

1. Provide a detailed description of the onsite storage facility for the replaced steam generator lower assemblies. Include a description of all systems designed to minimize the spread of radioactivity (e.g., floors, sumps, air filtration units).

RESPONSE

Preliminary engineering design consists of a 60 ft by 110 ft steam generator storage facility designed as a reinforced concrete structure, with a watertight concrete roof.

Since the steam generators will be seal welded in addition to being in a facility having a watertight concrete roof and reinforced concrete walls, there are no potential means to transport the surface contamination from the lower assembly surfaces. Therefore a floor, sumps and/or air filtration units are not required.

It is anticipated that the steam generator storage facility will be located in the laydown area approximately 150 feet south of the ash disposal pits and 350 feet east of the Radwaste Building, or approximate area, per Figure D.1-1.

The design of the wall thickness was determined using a point-kernal computer code which used semi-empirical methods developed by Rockwell[1] for calculating the direct gamma dose rates from a homogeneous volumetric cylindrical source through slab shields.

The values of the source terms for the analysis were based on the results of a field survey* of a steam generator in a drained condition one month after shutdown. For conservatism it was assumed that all short-lived isotopes had decayed away and the sole contributor to the measured dose rate was cobalt-60, which has the highest average gamma-ray energies and is therefore the most difficult to shield for a given curie level. The results of the conservative analyses indicate that 24-inch concrete walls are required to meet the dose criteria of 2.5 mr/hr at the exterior wall surfaces. However, the dose at the exterior wall surface is expected to be at or below 0.25 mr/hr.

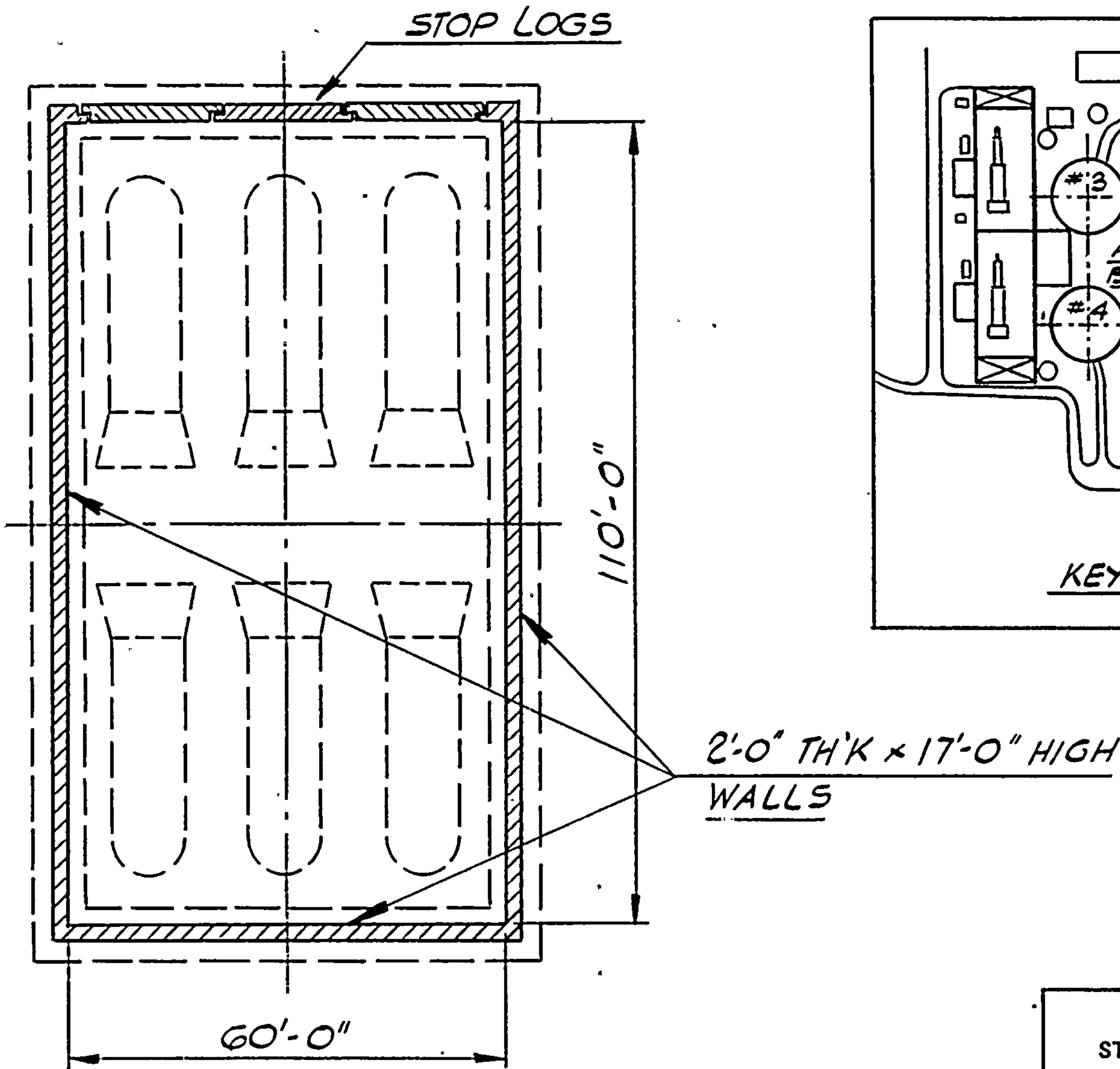
The skyshine analysis was performed with an industry-recognized computer code G^3[2] based upon the same field survey* of a steam generator, assuming the average energy of Co^{60} as that of the source. The source strength of the isotropic point source was determined by calculating a normalization constant equal to the total photon leakage from the steam generator. The skyshine contribution, without taking credit for a shielding roof, will not increase the dose rate outside the compound over 0.25 mr/hr.

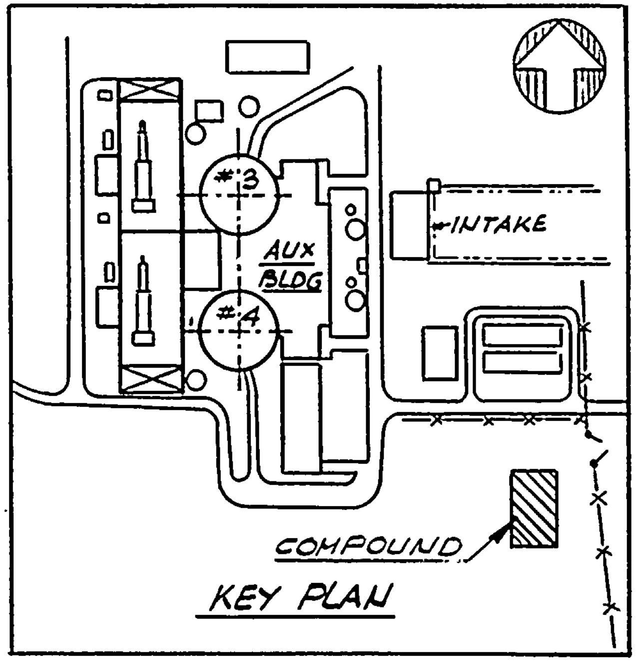
*See SGRR Figure 3.3-7

(1) T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., New York (1956).

(2) R. E. Malefant, G^3: A General Purpose Gamma-Ray Scattering Program, Los Alamos Scientific Laboratory, LA 5176 (June 1973).

The resulting dose equivalent to an individual at the north site boundary location for a full year was calculated assuming 2.5 mr/hr at the outside surfaces of the storage compound, plus the skyshine contribution assuming no roof on the storage facility. The calculated dose was 5.2 X 10^{-3} mrem which is considered an insignificant contribution of the offsite dose (see SGRR Section 3.4.6). The presently proposed facility location (see Figure D.1-1) was assumed for the aforementioned analysis.





Note: These dimensions are preliminary and may change with final design.

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

PROPOSED
STEAM GENERATOR STORAGE COMPOUND
FIGURE D.1-1 Revision 6
January 1979

6. It is the staff's position that significant amounts of Fe-55 and Ni-63 may be released during the steam generator repair effort. It is our position that you (1) commit to perform a monthly composite sample of all liquid effluents for Fe-55 and Ni-63 or (2) provide adequate justification for not monitoring for these isotopes.

RESPONSE

Fe-55 and Ni-63 are both low energy emitters which require complex techniques to determine isotopic composition. The gross beta activity is reported in our semi-annual report. The bio-accumulation factor (NRC Reg. Guide 1.109) for these isotopes is not well established. Consequently we do not feel that performing a monthly composite sample of all liquid effluents for Fe-55 and Ni-63 is cost justifiable. However, at least one representative laundry sample will be checked for Fe-55 and Ni-63 during the steam generator repair effort (obtained during the period when RCS pipe cuts are made) to determine the magnitude of releases.